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DRAFT VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

All NRC licensees are required to provide appropriate radiation protection training for all permanent and transient personnel who work in restricted areas (§19.12 of 10 CFR Part 19). A clear and reasonable assessment of the biological risks associated with occupational radiation exposure is essential to effective radiation protection training. The proposed action is to provide instructional material in a suitable form describing and estimating the risks from exposure to radiation. The instructional material will be suitable for use in licensee training programs and will represent an acceptable method of complying with part of the existing training requirements.

1.2 Need for Proposed Action

One common element of those occupational areas encompassed by NRC licensing activity is worker exposure to ionizing radiation and the biological risks from exposure. Union representatives have expressed a dissatisfaction with the way in which these risks have been explained to the worker by the licensee. In addition, they feel the NRC has a responsibility to make its position on the controversial issue of radiation risk clear to the worker and the public. A meeting of NRC staff and union representatives was held on November 28, 1978, during which this matter was discussed. A transcript of the meeting is available from the Public Document Room.

The Commission has directed the staff to prepare for and initiate a public hearing concerning the adequacy of present occupational radiation protection standards for exposure of individuals. This hearing should help resolve existing uncertainties in this complex area and the findings should, as a minimum, be published in a form suitable for instruction of the worker. Work on this project began prior to the public hearings so that updated information on risk could be disseminated to the worker shortly after the hearing. Most of the questions of concern to the unions can be disseminated to the worker shortly after the hearing. Most of the questions of concern to the unions can be answered now.

1.3 Value/Impact of Proposed Action

1.3.1 NRC Operations

Instructional material on radiation risk written at a level and scope understandable to the worker should contribute to increased confidence, on the part of the worker, in the NRC in general. A better understanding of the risk should elicit more worker cooperation with NRC-enforced safety programs. Impacts of the development of instructional material on risk are task completion manpower cost, estimated to be 0.2 man-year and printing costs of approximately $400.00.

1.3.2 Other Government Agencies

Agreement States whose licensing regulations include radiation protection training requirements may benefit from the availability of an NRC guide on radiation risk suitable for inclusion in those training programs. Development of the risk guide entails coordination with the Environmental Protection Agency, the Occupational Safety and Health Administration, and the Bureau of Radiological Health to avoid inconsistencies.

1.3.3 Industry

Providing a reasonable and understandable statement on worker risk should facilitate industry efforts to provide effective safety training and to better achieve as low as is reasonably achievable (ALARA) objectives. Minimal impact is expected in the form of additional cost of training programs since training requirements already exist. Input from unions and industry in the development of instructional material on risk will be encouraged, and this implies some additional costs such as staff time for reviewing drafts.

1.3.4 Workers

The proposed action should improve worker protection in that reasonable understanding of radiation risk is essential to the development of safe working practices. The staff believes that an objective discussion of radiation risk may in fact reduce "over concern" on the part of some workers. If improved training results in a wider recognition and respect for radiation as an industrial hazard, more attention will be given to protective procedures and a reduction in individual and collective dose should result.

1.3.5 Public

Nuclear workers are also members of the public and are generally residents of the area where facilities are located. Having a better informed public should result in a wider range of input to local decisionmaking concerning nuclear development. Improved training implies the added benefit of increased plant safety, thereby decreasing the probability of accidents that could involve the public.

1.3.6 Decision on Proposed Action

The NRC should develop and provide instructional material concerning risk from occupational radiation exposure.

2. TECHNICAL APPROACH

The technical approach proposed is to develop instructional material concerning risks to the worker from occupational radiation exposure and to publish the material in a form that will receive the widest dissemination among NRC-licensed facilities. An alternative is to publish the findings of the proposed hearing on dose limits and assume the relevant information will filter down to the worker. It is the feeling of the staff that a direct approach is required here.

3. PROCEDURAL APPROACH

The proposed action, to publish training material concerning risks from occupational radiation exposure, the use of which would be required of all licensees, could be accomplished by several alternative methods. These include an NRC regulation requiring that specific training materials be used, a regulatory guide based on the existing §19.12 that would provide an acceptable method for training on risks, an ANSI standard on training that could be adopted by a regulatory guide, and a NUREG report or a branch position paper.

3.1 Value/Impact of Procedural Alternatives

An NRC regulation establishes general legal requirements, is costly and time consuming to prepare, and is not an appropriate vehicle for the specific and narrow objective proposed here. A regulation would be difficult to modify as new information on radiation risk is developed. One advantage is that a regulation legally requires compliance. In general, this approach is not considered cost effective in view of the objectives of the proposed action.

ANSI standards are generally intended as highly technical and advanced treatments of specialized areas of concern to industry. A comprehensive technical review of risks from radiation would be of value but would not be suitable as instructional material at an introductory level for worker radiation protection training. Completion of an ANSI standard and an endorsing regulatory guide would require several years and would be too costly. This approach is not considered cost effective in view of the proposed objectives.

A NUREG document would be an appropriate vehicle for a comprehensive discussion of radiation risk beyond the scope of what is proposed here. A regulatory position, however, is not established through publication of a NUREG report. Since this proposal includes establishing an acceptable method for compliance with elements of required training programs, a NUREG report is not suitable.

Branch position statements are intended as interim measures to be used when an immediate response is required. They are usually superseded when a more permanent mode of guidance is developed.

A regulatory guide can be prepared at reasonable cost within a reasonable time period. The staff does not consider that revision of any existing regulatory guides could provide the instructional material intended here. Regulatory guides on training requirements are being developed but are specific to types of licensees such as Draft Regulatory Guide OH 717-4 for LWRs. The action proposed here has broad application to all licensees, as does Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure."

3.2 Decision on Procedural Approach

The staff concludes that work should begin on a regulatory guide similar to Regulatory Guide 8.13 on the subject of worker instruction concerning risks from occupational radiation exposure. Publication of the active guide should not occur until public hearings on the question of dose limits and risks have been held.

4. STATUTORY CONSIDERATIONS

4.1 NRC Regulatory Authority

Section 19.12 of 10 CFR Part 19 establishes a legal requirement that all NRC licensees provide radiation protection training to personnel and that the training be commensurate with the potential risks from radiation exposure encountered by those personnel. The NRC is thus authorized to provide criteria for acceptable levels of training and to inspect for compliance with training requirements.

4.2 Need for NEPA Statement

The action proposed here is to publish an instructional document on risks. This would occur after, and be in addition to, any major NRC action on retaining or modifying existing dose limits, based on planned public hearings. Since at that time it would not constitute a major addition or change and would entail no effect on the environment, an environmental impact statement is not considered necessary.

5. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICIES

Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," will require a commitment to appropriate radiation protection training. When next revised, it should include reference to this proposed action as an acceptable element of a licensee's training program.

This proposed guide is consistent with Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable." When next revised, it should include cross-reference to this proposed action.

This proposed action directly supplements the Draft Regulatory Guide OH 717-4, "Radiation Protection Training for Light-Water-Cooled Nuclear Power Plant Personnel," and will supplement and be referenced in other planned guides on training at other types of licensed facilities, e.g., uranium fuel fabrication plants, uranium mills, medical institutions.

6. SUMMARY AND CONCLUSIONS

In summary, it is proposed that a regulatory guide be prepared and issued for the purpose of providing instructional material concerning an assessment of risk from occupational radiation exposure.

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U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF STANDARDS DEVELOPMENT
DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

May 1980
Division 8
Task OH 015-4

Contact: S. A. McGuire (301) 443-5970

PROPOSED REVISION 1 TO REGULATORY GUIDE 8.12
CRITICALITY ACCIDENT ALARM SYSTEMS

FOR COMMENT

A. INTRODUCTION

Section 70.24, "Criticality Accident Requirements," of 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," requires licensees who may possess special nuclear material in excess of certain amounts to maintain a criticality accident alarm system. This guide describes a system acceptable to the NRC staff for meeting the Commission's requirements for a criticality accident alarm system.

B. DISCUSSION

Paragraph 70.24(a)(1) requires, in part, the detection of a criticality that produces an absorbed dose in soft tissue of 20 rads of combined neutron and gamma radiation at an unshielded distance of 2 meters from the reacting material within 1 minute. This sensitivity should be enough to detect any accidental criticality. Criticality accident alarm systems are also discussed in American National Standard ANSI/ANS-8.3-1979,[1] "Criticality Accident Alarm System," which has the same detection criterion. *

C. REGULATORY POSITION

The guidance on criticality accident alarm systems contained in ANSI/ANS-8.3-1979, "Criticality Accident Alarm System," is generally acceptable, subject to the following:

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* Lines indicate substantive change from previous issue.

[1] Copies may be obtained from the American National Standards Institute, Inc., 1430 Broadway, New York, New York 10018.

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by JUL 25 1980

Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

1. Section 70.24 requires alarm coverage "in each area in which such licensed special nuclear material is handled, used, or stored...," instead of "wherever it is deemed that they will result in a reduction in total risk," as in paragraph 4.1 of the standard. The NRC staff recognizes that some areas where special nuclear material is present may always have quantities that are too small to go critical. Examples of such areas could be sampling laboratories or areas having a neutron source using special nuclear material. The ANSI standard discusses the conditions where criticality is not considered possible or where doses may be small enough that alarm coverage is not needed. Licensees may apply for an exemption from the alarm requirement for these areas. Requests for exemptions should show that critical quantities of special nuclear material will never be found in these areas.

2. Paragraph 70.24(a)(1) requires that each area be covered by two detectors, in contrast to paragraph 4.5.1 of the standard that allows coverage by a single detector.

3. A quick screening capability as required in paragraph 70.24(b) may be provided instead of the more extensive dosimetry requirements in paragraph 6.6 of the standard.

D. IMPLEMENTATON

This proposed Revision 1 of the guide has been released to encourage public participation in its development. Except in those cases in which an applicant has proposed an acceptable alternative method for complying with specified portions of the Commission's regulations, the methods described in the currently active guide dated December 1974 will continue to be used in the evaluation of license applications docketed after February 28, 1975, for a license to possess special nuclear material in quantities subject to § 70.24. The methods to be described in the active Revision 1 reflecting public comments will be used in the evaluation of all license applications docketed after the implementation date to be specified. The implementation date will in no case be earlier than January 1, 1981.

DRAFT VALUE/IMPACT STATEMENT

This revision of Regulatory Guide 8.12, "Criticality Accident Alarm Systems," proposes a relatively minor change to incorporate by reference a newly revised ANSI standard, ANSI/ANS-8.3-1979, "Criticality Accident Alarm System."

The standard is not significantly changed from the earlier standard, and the proposed revision does not represent a change in NRC position. Therefore, neither the value nor the impact of the guide would change with the proposed revision.

The reason for seeking public comment at this time is not that NRC is proposing a new policy, but rather to seek comment on the existing policy. Such comment is appropriate at this time since the original guide was last published over 5 years ago and since the revised ANSI standard has just been published.

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DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

June 1980
Division 10
Task TP 602-4

Contact: M.E. Wengler (301) 443-5946

PROPOSED REVISION 1* TO REGULATORY GUIDE 10.6

GUIDE FOR THE PREPARATION OF APPLICATIONS FOR USE OF SEALED SOURCES AND DEVICES FOR PERFORMING INDUSTRIAL RADIOGRAPHY

FOR COMMENT

1. INTRODUCTION

1.1 Purpose of Guide

The purpose of this guide is to provide assistance in the preparation of an application for a Nuclear Regulatory Commission (NRC) license for possession and use of sealed sources and devices for performing industrial radiography. "Radiography," as used in this guide, means the examination of the structure of materials by nondestructive methods utilizing sealed sources of byproduct materials (radioisotopes). The radioisotopes most commonly used for radiography are cobalt-60 and iridium-192.

1.2 Applicable Regulations

Commission regulations that apply to radiography and that should be used in conjunction with this guide are 10 CFR Part 19, "Notices, Instructions, and Reports to Workers; Inspections;" 10 CFR Part 20, "Standards for Protection Against Radiation;" 10 CFR Part 21, "Reporting of Defects and Noncompliance;" 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material;" 10 CFR Part 34, "Licenses for Radiography and Radiation

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* The substantial number of changes in this proposed revision has made it impractical to indicate the changes with lines in the margin.

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by AUG 29 1980

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Safety Requirements for Radiographic Operations;" 10 CFR Part 40, "Domestic Licensing of Source Material;" 10 CFR Part 71, "Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions;" and 10 CFR Part 170, "Fees for Facilities and Materials Licenses and other Regulatory Services Under the Atomic Energy Act of 1954, as Amended." The applicant should carefully read these regulations and this guide and should submit all information requested.

1.3 As Low As Is Reasonably Achievable (ALARA)

Paragraph 20.1(c) of 10 CFR Part 20 states, in part, "...persons engaged in activities under licenses issued by the Nuclear Regulatory Commission pursuant to the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974 should, in addition to complying with the requirements set forth in this part, make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to unrestricted areas, as low as is reasonably achievable." Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," provides the NRC staff position on this important subject. License applicants should give consideration to the ALARA philosophy, as described in Regulatory Guide 8.10, in the development of plans for work with licensed radioactive materials.

2. LICENSE FEES

A license fee is required for most types of licenses and amendments to licenses. The applicant should refer to §170.31, "Schedule of fees for materials licenses and other regulatory services," of 10 CFR Part 170 to determine the amount of fee that must accompany the application. Review of the application will not begin until the proper fee is received by the NRC.

3. FILING AN APPLICATION

An application for a license should be filed on Form NRC-313R, "Application for Byproduct Material License - Use of Sealed Sources in Radiography," (Exhibit A) and should contain all the information specified in the application form. The space provided on the form is limited; therefore, the applicant should append additional sheets to provide complete information.

The application should be completed in triplicate. The original and one copy should be mailed to the Materials Licensing Branch, Division of Fuel Cycle and Material Safety, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. One copy of the application with all attachments should be retained by the applicant, since the license will require, as a condition, that the licensee follow the statements and representations set forth in the application and any supplements to it.

4. CONTENTS OF AN APPLICATION

The following discussion deals with specific items on the application form. The information specified in Items 1 through 3 of the application is explained in the instruction sheet accompanying the application form, and Item 7 is self-explanatory. The information needed for Items 4, 5, and 6 of the application form is discussed below. In completing the application, especially Items 4 and 5, the applicant should cover anticipated expansion and changes in the program and thus reduce the need for future license amendments.

Item 4. The sealed sources including calibration sources that the applicant will possess and use should be listed by radioisotope, manufacturer, and model number. The maximum amount of radioactivity in each source should be specified. The number of sources that the applicant will possess at any one time need not be specifically stated unless any one source contains more than 100 curies of Cobalt 60.

Item 5. Radiographic exposure devices should be designated by manufacturer and model number and should be keyed alphabetically to the sources listed in Item 4 with which they will be used (specify in 5(a) of the application form).

If source changers will be used, the source changers should be identified by manufacturer and model number and should be keyed alphabetically to the source-device combinations with which they will be used (specify in 5(b) of the application form).

If necessary, the suppliers of equipment should be contacted concerning the model numbers of sources, devices, and source changers to ensure that the information contained in the application is accurate. Improperly identified equipment may require additional correspondence.

Item 6(a). This item applies only to permanent radiographic installations defined in Paragraph 34.2(h) of 10 CFR Part 34. If a permanent radiographic installation will be used for performance of radiography, a detailed description of the installation that includes the following information should be submitted:

a. Annotated drawings or sketches of the installation and its surroundings, including (1) dimensions of each enclosed area; (2) thickness, density, and type of shielding material on all sides, above and below; (3) identification of entranceways; and (4) a description of the nature of and distances to all areas adjacent to, above, and below each exposure area.

b. A description of the area security safeguards, such as locks, signs, warning lights and alarms, and interlocking systems for each enclosed exposure area and adjacent areas. Particular attention should be given to the description of the high radiation area entrance controls that are required by § 34.29. The applicant should describe the means for ensuring that the interlock systems required by § 34.29 are not easily defeated by the radiographer. This may be accomplished through system design, e.g., replacement of simple switches by direct wiring or key switches under control of the radiation safety officer, or through strict control procedures, e.g., direct supervision of the operation by the radiation safety officer.

c. The results of calculations or radiation level measurements showing maximum anticipated radiation levels in all areas adjacent to each exposure area including the roof or ceiling. As a basis for calculations, the type of source, quantity of activity in the source, and position of the source within the installation should be identified.

Particular attention should be given to radiation levels on the roof of the installation. If those levels may exceed 2 milliroentgens per hour, the application should show how access to the roof will be controlled. If the calculations or measurements show that radiation levels on the roof might exceed 100 milliroentgens per hour, the applicant should consider the use of collimating devices or additional shielding in the roof or ceiling.

A properly shielded installation will permit the performance of radiography within the facility with the areas outside the facility considered as unrestricted areas if they meet the radiation level limitations in paragraph 20.105(b) of 10 CFR Part 20. A radiation level of not more than 2 milliroentgens per hour at a distance of 18 inches from any external surface of the facility will be considered acceptable for considering the area as an unrestricted area.

Item 6(b). Instruments should be identified by type (i.e., ionization, G.M., scintillation) and exposure range. Instruments to be used for surveys are required by § 34.24 to measure from a minimum of two milliroentgens per hour through one roentgen per hour. The manufacturer and model number of the survey instrument should be identified.

Item 6(c). Section 34.24 requires that radiation survey instruments used in radiographic operations be calibrated at intervals not to exceed 3 months and after each instrument servicing. Appendix A contains a description of an acceptable procedure for calibrating survey instruments.

If an applicant wishes to calibrate his or her own instruments, the following information should be submitted:

a. The type (radioisotope, manufacturer, and model number) and activity of any source to be used for calibration.

b. The accuracy* of the source.

c. The specific procedures to be used for calibration including radiation safety procedures to be followed for use of the source.

d. The name and pertinent experience of each individual who will perform instrument calibration. This information may be omitted if the individual is a radiographer and the applicant has an adequate program for the training of radiographers (see 6(f)) that includes instrument calibration procedures. If radiographers will perform instrument calibration, specific instructions and procedures should be written for use by radiographers and should be included in the operating and emergency procedures (see 6(e)).

If instrument calibration will be performed by an organization other than the applicant, the application should include (1) the name and address of the organization, (2) a description of the calibration procedure, and (3) a description of the instrument calibration data that will be furnished to the applicant.

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*Accuracy is the maximum deviation of the nominal value of the source from the true value. This information is normally provided by the manufacturer.

Item 6(d). The applicant should specify the type of personnel monitoring device (for example, film badge or thermoluminescent dosimeter) that will be used, and the names and addresses of the suppliers that may be used. The frequency of change of film badges or thermoluminescent dosimeters should be specified.

The manufacturer, model number, and range of pocket dosimeters to be used should be identified. Section 34.33 of 10 CFR Part 34 requires that pocket dosimeters have a range from zero to at least 200 milliroentgens. Dosimeters with an extremely high range, for example, dosimeters designed for Civil Defense purposes, which have a range from zero to 100 roentgens, are difficult to read in the zero to 200 milliroentgens range and are not considered acceptable. Procedures for checking pocket dosimeter exposure and energy response, as required by paragraph 34.33(c), should be described.

Item 6(e). The applicant should describe the operating and emergency procedures that will be followed by radiography personnel. Appendix B describes items and procedures that should be included.

Item 6(f). The applicant should provide a description of the complete training program for radiographers, radiographers' assistants, and experienced radiographers and radiographers' assistants. This description should include appropriate references to any instructions given by outside service agencies. The name, training, and experience with radiation of each person who is to participate in the instruction, examination, or qualification of trainees should be given in sufficient detail to establish his or her qualifications to perform these services. If an individual will teach only certain parts of the course, this should be specified. Appendix C describes the elements of an acceptable training program.

In certain cases where a radiography program is limited to a few individuals and personnel turnover is not anticipated, a training program may not be necessary provided each individual has had adequate training and experience. In such a situation, the application should request that specific individuals be named on the license in lieu of submitting details of a complete training program. The application should explain why it is not necessary to establish a training program. The qualifications of each individual designated as a radiographer or radiographer's assistant should be described and should include the following information:

a. The individual's name and the capacity in which that individual will function (radiographer or radiographer's assistant).

b. A detailed description of each individual's training and experience in the principles of radiation and radiation safety. The information provided should specify when (dates) and where training was received. The name of the individual who provided the training should be indicated if training other than that of a commercially offered program was used.

c. The specific experience of each individual in the use and handling of the type of equipment specified in the application. The information should include names of previous employers, dates employed, type of equipment used, and length of time the equipment was used for each previous employer.

d. A description of the instruction that each prospective radiographer and radiographer's assistant has received in the applicant's operating and emergency procedures.

e. A description of the means used to determine competence of individuals to act as radiographers and by whom such determination is made. Copies of examinations given to determine knowledge and understanding of the topics in Appendix A of Part 34, the applicant's operating and emergency procedures, Commission regulations, and use of equipment should be submitted.

f. A description of the means used to determine competence of individuals to act as radiographers' assistants and by whom such determination is made. Copies of examinations given to determine understanding of the licensee's operating and emergency procedures and means used to determine competence to use radiographic and related equipment should be submitted.

g. A description of the periodic training program for personnel.

The limitations imposed in licenses when the applicant has not established a complete program for training radiographers and radiographers' assistants should be clearly understood. The license will specifically name each individual authorized to act as a radiographer or radiographer's assistant. Should all radiographers who are named in the license become unavailable, the licensee would not be permitted to perform radiography under the license and therefore would have, in effect, authorization to possess the licensed byproduct material for storage purposes only. The individuals named on the license are required to perform all radiographic operations conducted under the license, including the manipulation of equipment and the performance of radiation surveys. Although application may be made for the addition or substitution of names of other individuals in the license, the time needed for preparation and submission of the application and for the Commission to issue an amendment could delay the program. No individual may act as a radiographer's assistant unless the license has been amended to provide for an individual to act in that capacity. An individual should not enter into on-the-job training until the license has been amended to permit that training.

Item 6(g). The applicant should provide a description of the internal inspection system for controlling the receipt, possession, and use of radioactive material. The description should show how the system ensures that license conditions, Commission regulations, and operating and emergency procedures are followed by radiographers and radiographers' assistants as required by Paragraph 34.11(d).

Other management controls, including a description of (1) the qualifications of each person responsible for maintaining such control, (2) the type of internal inspections to be made and their frequency, (3) the responsibilities of each person in the program, (4) the procedure for recording and reporting deficiencies to appropriate management personnel, and (5) the education and followup program to be used in correcting deficiencies noted during inspections should be specified in the application. The type and extent of the radiography program to be conducted will usually determine the nature of the system and the inspection frequency. Paragraph 34.11(d) requires that an internal inspection be performed at intervals not exceeding three months and that records of such inspection be retained for two years.

Internal inspections, including evaluation of each radiographer, should be made by a person of authority in management. These inspections may be announced or unannounced as necessary to ensure compliance with Part 34. This person should have a thorough knowledge of equipment, procedures, and regulations and level of competence at or above that expected of a radiographer. Management should make a continuing review of quarterly inventories, utilization logs, records of receipt and disposal of licensed material, personnel monitoring results, and survey.

Item 6(h). Paragraph 34.11(e) requires that the license applicant submit a description of the overall organizational structure pertaining to the radiography program, including specific delegations of authority and responsibility for the program. The applicant should describe how active control over the radiography program is exercised by management personnel in positions of authority. Each individual in the line of authority should be identified by name and title. In addition, the applicant should provide the name, training, and experience of that individual in management who will be assigned duties established by the licensee for maintaining an active management control of the radiation program and radiographic operations. Appendix D describes acceptable qualifications for such individuals and the responsibilities of the positions. If persons of lesser authority will assume some of the duties and responsibilities normally reserved for management, the application should identify those persons and specify how management will ensure that their duties are properly performed.

Item 6(i). The applicant should submit a description of the leak testing program for the sealed sources as follows:

a. If the services of a consultant or commercial organization licensed by the Commission to take the necessary test samples (smears), evaluate the samples, and report the results to the customer are used, the name, address, and license number of the consultant or commercial organization should be specified.

b. If the applicant wishes to be licensed by the Commission to use a commercially available leak-test kit, the application should identify each kit to be used by designating the kit supplier and the kit model number. Only leak-test kits that are identified will be authorized. The application should also identify the individuals who will perform the leak test (using the kit).

c. If an applicant wishes to be licensed by the Commission to perform leak tests, including taking and evaluating the smears, paragraph 34.11(f) requires the applicant to describe the procedures to be used. The following information should be included:

(1) a description of the instrumentation to be used in evaluating the smears, including its sensitivity and accuracy;

(2) a description of the calibrating and standardizing procedures with a sample calculation showing conversion of results to the required microcurie units. Survey instruments are generally not designed for such measurements and may not be acceptable for this use;

(3) a description of the material to be used in taking the smears, the points on the equipment that will be smeared (smears are not normally taken directly from the surface of a source; see paragraph 34.25(c));

(4) the radiation safety procedures to be followed during the smearing process, the method for handling and disposing of the smears; and

(5) a description of the pertinent training and experience of each person who will take or evaluate the smears.

Distributors of sealed sources usually supply a certificate with each source giving the results and date of the last leak test performed on a source. If such a certificate is not received, the source may not be used until a leak test has been performed and the results of the test showing that the source is not leaking or contaminated have been received. Thereafter, the source must be tested for leakage and contamination at intervals not to exceed 6 months. Records of the testing identifying each source tested, the date of the test, and the results of the test in units of microcuries must be maintained for Commission inspection.

5. AMENDMENTS TO LICENSES

Licensees are required to conduct their programs in accordance with statements, representations, and procedures contained in the license application and supportive documents and in the conditions of the license. The license must therefore be amended if the licensee plans to make any changes in facilities, equipment (including monitoring and survey instruments), procedures, personnel, or byproduct material to be used.

Applications for license amendments may be filed either on the application form or in letter form. The application should identify the license by number and should clearly describe the exact nature of the requested changes, additions, or deletions. References to previously submitted information and documents should be clear and specific and should identify the pertinent information by date, page, and paragraph.

The application for license amendment must be accompanied by the proper fee (see 10 CFR § 170.31).

6. RENEWAL OF A LICENSE

An application for renewal of a license should be filed at least 30 days prior to the expiration date as provided for in paragraph 30.37(b) of 10 CFR Part 30. This will ensure that the license does not expire until final action on the application has been taken by the NRC.

Renewal applications should be filed on Form NRC-313R appropriately supplemented and should contain complete and up-to-date information about the applicant's current program. The renewal application must be accompanied by the proper fee (see 10 CFR § 170.31).

In order to facilitate the review process, the application for renewal should be submitted without reference to previously submitted documents and information. If such references cannot be avoided, they should be clear and specific and should identify the pertinent information by date, page, and paragraph.

APPENDIX A

CALIBRATION OF INSTRUMENTS

1. Calibration of survey meters should be performed with radionuclide sources* that approximate point sources.

a. The source activities should be traceable within 5% accuracy to the U.S. National Bureau of Standards (NBS calibrations).

b. Section 34.24 requires that survey instruments be calibrated at least quarterly and after servicing.

c. Each scale in the range from two milliroentgens per hour through one roentgen per hour is required to be calibrated. All others should also be calibrated.

d. The highest and lowest points used to calibrate each scale of the instrument should be separated by at least 50% of the scale.

e. The exposure rate measured by the instrument should differ from the true exposure rate by less than 20% of the values specified for the standard source.

f. A calibration chart or graph containing at least the date of last calibration and due date of next calibration should be affixed to the survey meter.

2. The use of the small check source that is incorporated into some survey meters is not appropriate or acceptable for calibration purposes.

3. Section 34.24 requires that records of the calibrations discussed in Item 1 be maintained.

4. The inverse square law and radioactive decay law may be used for calibration. A calibrated source will have a calibration certificate giving its output at a given distance measured on a specified date by the manufacturer. The inverse square law may be used with any point source to calculate the exposure rate at other distances. The radioactive decay law may be used to calculate the output at any time after the specified date.

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* Sources of cesium-137, radium-226, or cobalt-60 are appropriate for use in calibrations. The activity of the calibration standard should be sufficient to calibrate the survey meters on all ranges up to one roentgen per hour. If there are higher ranges, they should at least be checked for operation and approximately correct response to radiation.

a. Inverse Square Law

If R_a is the exposure rate at a distance D_a from a point source and R_b is the exposure rate at a distance D_b from the same point source, then

$$R_a D_a^2 = R_b D_b^2$$

Note: R_a and R_b must be in the same units of exposure rate (e.g., mR/hour, R/hour, etc.) and D_a and D_b must be in the same units of distance (e.g., centimeters, meters, etc.).

If R_a, D_a, and D_b are known, R_b can be calculated from

$$R_b = \frac{D_a^2}{D_b^2} \times R_a$$

b. Radioactive Decay Law

The exposure rate of a standard source at a time after a specified calibration date is given by

$$R_t = R_o \times e^{-(0.693\frac{t}{T_{1/2}})}$$

where

R_t is the exposure rate at a time t after the source calibration date

R_o is the exposure rate on the day of calibration

t is the time elapsed since the calibration date

$T_{1/2}$ is the radionuclide half-life

Note: R_t and R_o must be in the same units of exposure rate (e.g., mR/hour, R/hour, etc.) and t and $T_{1/2}$ must be in the same units of time (e.g., seconds, days, years, etc.).

APPENDIX B

OPERATING AND EMERGENCY PROCEDURES

Section 34.31 of 10 CFR Part 34 requires each licensee to provide radiography personnel with operating and emergency procedures. The purpose of this requirement is to provide radiography personnel with clear and specific instructions in the topics listed in § 34.32 and in other duties and responsibilities that radiography personnel may have. Other duties could include instrument calibration, leak testing, quarterly inspection and preventive maintenance of equipment, and shipment of sources and devices. The operating and emergency procedures for personnel should not contain information that does not apply specifically to the duties of radiography personnel; for example, neither the training program nor the management control program should be included in the operating and emergency procedures.

The operating and emergency procedures should be tailored to fit the program proposed in the application. The procedures and instructions should be complete and self-contained in a single document or in a clearly designated part of a broader scope document that includes related material such as a description of the management control program. Information contained in equipment manuals and other publications should be extracted and inserted into the operating and emergency procedures so that the instructions to personnel are clear, specific, and appropriate for the proposed program. The instructions contained in the operating and emergency procedures should be in language that can be easily understood by radiography personnel. Where applicable, instructions for use and handling of devices incorporated into permanent radiographic installations should be separate and distinct from those for mobile or portable devices.

There is no specific format for operating and emergency procedures. However, a sequential set of instructions that covers radiography operations from the beginning of the workday to the end of the workday is an acceptable format. Topics that should be included in the operating and emergency procedures are:

a. Handling and Use of Licensed Sealed Sources, Radiographic Exposure Devices, Source Exchangers, and Instrument Calibration Equipment.

(1) Step-by-step instructions of the "cookbook" type for the use and handling of radiographic exposure devices and related equipment should be

provided. When appropriate, the procedures should include instructions for use of radiation collimating cones or other auxiliary shielding material.

(2) If source exchange will be performed by radiography personnel, step-by-step instructions for source exchange, including surveys to be performed during the source exchange and for shipment and acceptable radiation levels for these surveys, should be in the procedures. Such instruction should also state the steps to be taken if the survey levels exceed acceptable limits.

(3) If radiography personnel will perform instrument calibration, step-by-step instructions should be in the procedures.

(4) If radiography personnel will perform leak testing of sealed sources, specific instructions for performing the leak test should be in the procedures. If the applicant will use commercially available leak-test kits, the instructions and procedures provided by the kit suppliers should be modified to the applicant's program. For example, many kit procedures indicate that the manufacturer of the source should be notified if a survey of the leak-test sample indicates a potentially leaking source. Instructions should indicate that management will be informed since dealing with suppliers is usually a management function.

b. Methods and Occasions for Conducting Radiation Surveys. The procedures should identify when surveys should be made, specifically what should be surveyed, acceptable radiation levels for the surveys, the steps to be taken if acceptable levels are exceeded, and records of survey results to be made. In general, a survey should be performed each time a source is manipulated or moved. Surveys that need to be performed include:

(1) Determination after each exposure that the source has returned to the safe storage position. The entire circumference of the radiographic device must be surveyed. If the radiographic exposure device has a source guide tube, the survey must include the guide tube.

(2) Determination of the boundary of the restricted area.

(3) Determination of radiation levels at external surfaces of storage facilities.

(4) Determination of radiation levels in and around vehicles used for transporting or storing sources and devices.

(5) Determination that the source is in a safe storage position prior to securing a radiographic exposure or storage container.

(6) Determination that containers prepared for shipment comply with the requirements in Department of Transportation regulations (10 mR/hr at 3 feet from any surface and 200 mR/hr at the surface of the container).

The acceptable radiation levels for surveys should be expressed in milliroentgens per hour.

Section 20.401 and paragraph 34.43(c) require that records be maintained of specific surveys. Judgment by NRC licensees is required for maintaining records of other surveys required by the regulations in which maintenance of the records is not specified in the regulations. NRC does not expect radiographers to record each and every reading taken during a survey. However, records should be complete enough to show clearly that proper surveys have been done.

c. Methods for Controlling Access to Radiographic Areas. Instructions for controlling access to radiographic areas should be specifically stated in the procedures.

The boundaries of restricted areas and high radiation areas are required to be posted. "Caution[1] Radiation Area" signs should be posted at the boundary of the restricted area, and "Caution[1] High Radiation Area" signs should be posted at the boundary of the high radiation area. High radiation area signs should not be used at the boundary of a restricted area; these signs should be used only at the boundary of a high radiation area.

Signs, by themselves, do not provide an adequate means of access control. For radiographic operations performed outside a permanent radiographic installation, instructions requiring surveillance of the area to prevent unauthorized persons from entering the area are necessary. For permanent radiographic installations, specific instructions concerning use of interlocking devices and systems, locking of the facility, security of keys, use of warning lights and alarms, etc., should be included in the procedures.

The instructions for control of access to permanent radiographic installations should be separate and distinct from the instructions for temporary site operations.

A specification of a radiation level of 2 milliroentgens per hour for the boundary of the restricted area and 100 milliroentgens per hour for the boundary of the high radiation area is acceptable. A physical survey with a survey meter should be performed to confirm the 2 milliroentgens per hour radiation level for the restricted area boundary after the source has been exposed. It is neither necessary nor desirable for a physical survey to be made to confirm the radiation level at the boundary of the high radiation area since such a survey could lead to unnecessary exposure of personnel.

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[1] Or Danger

d. Methods and Occasions for Locking and Securing Radiographic Exposure Devices, Storage Containers, and Sealed Sources. Instructions requiring securing of the exposure device after completion of the survey to determine that the source has returned to the safe storage position should be written and available. Instructions should further specify that surveys to determine that the sealed source has returned to the shielded position after an exposure are required pursuant to § 34.43 and, in the case of a radiographer's assistant, they must be made under the personal supervision of a radiographer (§ 34.44). Section 34.43 requires that a record of the last survey made prior to locking the radiographic exposure device and ending direct surveillance be maintained for two years.

Instructions and procedures for storage of sources and devices at both permanent and temporary job sites, including locking of devices, posting of storage areas, and surveys around the storage areas, should be in the procedures. The area outside storage areas should meet the requirements for an unrestricted area.

e. Personnel Monitoring and the Use of Personnel Monitoring Equipment. The instructions should contain requirements for radiography personnel to wear their personnel monitoring devices so that any exposure received will be accurately reflected by the devices. The instructions should be specific.

Frequent reading of pocket dosimeters should be required so that personnel may be aware of exposure that they may have received. An instruction concerning steps that must be taken immediately by radiography personnel in the event a dosimeter is found to be off scale should be included in the procedures. This instruction should include the requirement stated in paragraph 34.33(c) that an individual's film badge or TLD be processed immediately if that individual's pocket dosimeter is discharged beyond its range. Instructions for storage of personnel monitoring devices should be included in the procedures.

f. Transporting Sealed Sources to Field Locations, Packaging of Exposure Devices and Storage Containers in the Vehicles, Posting of Vehicles, and Control of Sealed Sources During Transportation. Most transport of radiography sources in exposure devices or storage containers over public roads is subject to the regulations of the Department of Transportation. These regulations cover, among other things, permissible radiation levels around and within a vehicle and placarding of the vehicle during transport. Even in those cases in which the Department of Transportation regulations are not applicable (such as intrastate transportation), Part 71 of the Nuclear Regulatory Commission's regulations requires conformance to the standards and requirements of the Department of Transportation.

The procedures should contain instructions on how exposure devices or storage containers will be secured within the transporting vehicle to prevent shifting within the vehicle. There should be instructions for placarding of the vehicle during transport. The Department of Transportation regulations require "RADIOACTIVE" placards on all four sides of the vehicle. There should be instructions for surveys in and around the vehicle. The radiation level in the passenger compartment should not exceed 2 milliroentgens per hour. Although it is not specifically required for transport, there are occasions when the vehicle should be considered an unrestricted area so that a specification of a radiation level of 2 milliroentgens per hour at a distance of 18 inches from any external surface of the vehicle should be provided.

When a vehicle is used for storage, i.e., when the sources are not being transported, the posting requirements in 10 CFR Part 20 are applicable, and that vehicle should therefore be posted with "Caution--Radioactive Material" signs. As noted above, the area outside a parked vehicle used for storage is an unrestricted area, and the radiation level at 18 inches from the surface of the vehicle should not exceed 2 milliroentgens per hour.

g. Minimizing Exposure of Persons in the Event of an Accident. Instructions to personnel should include procedures for minimizing the exposure of persons in the event of an accident or other unusual occurrence. Possible malfunctions of equipment should be considered, and steps to follow in each case of malfunction should be specifically set forth.

The procedures should contain clear and specific instructions concerning emergency situations. The steps to be taken by radiography personnel should, in general, be limited to (1) surveying the area, (2) establishing a restricted area, (3) notifying appropriate persons, and (4) maintaining direct surveillance and control over the area until the situation is corrected. Limitations on action that may be taken by radiography personnel should be clearly specified. The attempted recovery of a source that has become detached from an exposure device, an operation that may result in exposure to high levels of radiation, should not be attempted by radiography personnel unless specifically trained.

h. Procedure for Notifying Proper Persons in the Event of an Accident. The names and telephone numbers of the persons to be contacted should be specified.

i. Recordkeeping. The instructions to personnel should specify those records that must be maintained by them during the course of their work. Among the records that are normally made by radiography personnel are dosimeter readings, surveys, and daily inspection of equipment. Other records should be included if they are the responsibility of radiography personnel. Records for which management and supervisory personnel have responsibility should not be included in the operating and emergency procedures.

j. Inspection and Maintenance of Radiography Exposure Devices, Storage Containers, and Source Changers. Section 34.28 requires a check for obvious defects in radiographic exposure devices, storage containers, and source changers prior to use each day the equipment is used. The procedures should contain specific instructions for inspection of equipment and the actions to be taken if any defects are found. A checklist should be contained in the procedures listing the items that should be covered in the daily inspection. Equipment manufacturers may be helpful in providing information concerning daily inspections.

Inspection and preventive maintenance of equipment at intervals not to exceed three months or prior to the first use thereafter are also required by § 34.28. If radiography personnel will conduct these inspections, the procedures should contain clear and specific instructions for inspection and maintenance. As part of the inspection and preventive maintenance program, all connectors, drive cables, source guide tubes, on-off indicator mechanisms, and moving parts should be checked for defects and excessive wear. Cables should be cleaned and lubricated, and all defective and excessively worn components repaired or replaced. If components esential to the safe operation of the device are found to be defective or in poor operating condition, the device should be immediately removed from service until repairs can be made. An instruction to be followed in this event should be written. Records of quarterly inspections and maintenance must be kept for two years.

Visible and audible warning systems used in a permanent radiographic installation should be tested and records made thereof at intervals not to exceed three months pursuant to § 34.28. Other area safeguards, which may include door and equipment interlocks or access-door locking devices, should be tested for proper operation at least once every six months. The procedures should contain instructions for performing such inspections if they are to be performed by radiography personnel.

k. Off-Scale Pocket Dosimeter Readings. Procedures to be taken immediately by radiography personnel in the event a pocket dosimeter is found to be offscale should include the following instructions:

(1) Stop work immediately;

(2) Initiate emergency procedures if the source is exposed and cannot be retracted; otherwise, retract the source safely;

(3) Notify the radiation safety officer immediately. In this regard the name of the radiation safety officer and the manner in which this individual can be reached should be included.

l. Product Malfunctions and Defects. If the radiographer discovers any malfunction or defect in the equipment, the radiographer should notify the radiation safety officer. Procedures to be followed in such an event should tell the radiographer what to report, when to report the problem, and the individual to whom it should be reported.

APPENDIX C

TRAINING PROGRAM

An applicant for a radiography license is required to have an adequate program for the training of radiographers and radiographers' assistants. With respect to radiography personnel, two important points should be understood: (1) the duties and responsibilities of the radiographer may not be delegated to the radiographer's assistant. A radiographer must be physically present at the location where radiography is being performed. A radiographer's assistant may perform source manipulation, surveys of the radiographic device to determine source location, etc., only in the physical presence of a radiographer, in accordance with § 34.44 and (2) any individual who assists a radiographer by manipulating radiographic exposure devices, sealed sources, related handling tools, or survey instruments is acting in the capacity of a radiographer's assistant and must meet the requirements of paragraph 34.31(b) of 10 CFR Part 34.

The description of the complete training program should include the sequence of events in the training of a person to become a radiographer or radiographer's assistant from the time of hiring through the time the job begins. Since 10 CFR Part 34 has different requirements for radiographers and radiographers' assistants, separate narratives pertaining to the training of individuals for each category should be submitted. In addition, a third narrative pertaining to the training of individuals who are hired with previous training and experience may be desirable. The narrative should include appropriate references to the more detailed description of each of the various parts of the training program described in accordance with the items identified below.

1. Initial Training

a. Radiographers

(1) Classroom Training. A description of the manner in which radiographer trainees will be instructed in all areas of Appendix A to 10 CFR Part 34 should be given. The course content should include a detailed outline of the topics in Appendix A to 10 CFR Part 34 and the approximate time to be spent on each major area of instruction should be specified. For individuals with no previous training, a minimum of 40 hours of classroom training should be provided in the subjects identified in Appendix A to Part 34.

A training course given by an outside service organization usually will not instruct the trainee with respect to an applicant's particular equipment, facilities, and procedures. Therefore, if outside training is to be used, instruction given to supplement that training with respect to the applicant's own equipment, facilities, and procedures and the method to be used to determine each trainee's competence in accordance with § 34.31 of 10 CFR Part 34 should be included in the program description.

(2) On-the-Job Training. A period of on-the-job training, including trainee observation of the use of radiographic exposure devices and associated equipment by experienced personnel and the trainee's use of such equipment under the supervision of experienced personnel, should be described. However, no individual should be permitted to enter into on-the-job training until completing at a minimum, the requirements for a radiographer's assistant. The content of on-the-job training and the minimum time that will be spent in it will be dictated by the applicant's scope of operation, the variety of the work, and the aptitude of the trainee and should be specified in the application. A minimum of three months full-time equivalent work as a radiographer's assistant should be provided.

(3) Examinations. A comprehensive examination containing, for example, 50 questions covering all items in Appendix A to 10 CFR Part 34 would be considered to be an adequate examination to qualify individuals as radiographers. In addition, a practical or field examination should be administered to determine the competency of radiographic trainees to perform surveys, posting, operate equipment, etc. Additional information is provided in Item 3, "Testing Procedures." Successful candidates should be given additional instruction in those areas where the examination indicated weakness.

b. Radiographers' Assistants

(1) Classroom Training. A description of the training program for radiographer assistant trainees should be given. Paragraph 34.31(b) specifies training requirements and the subjects in which radiographers' assistants must demonstrate understanding and competence. The description should contain a detailed outline of those items covered in the operating and emergency procedures and the use of radiographic equipment and should specify the approximate time to be spent on each major item listed in the outline.

(2) Examinations. A comprehensive examination containing, for example, 25 questions covering the applicant's operating and emergency procedures and the use of radiographic equipment would be considered to be an adequate examination to qualify individuals as radiographers' assistants. Additional information is provided in Item 3, "Testing Procedures." Successful candidates should be given additional instruction in those areas where the examination indicated weakness.

c. Experienced Radiographers and Radiographers' Assistants

Because of differences in procedures, equipment, etc., it is unlikely that a new employee will be adequately prepared to work in a particular program without some training specifically related to that program. Also each licensee is required by § 34.31 to make a determination that each individual is qualified to act as a radiographer or radiographer's assistant in its own specific program. The applicant should therefore describe the procedure for determining the knowledge and competency of such individuals and for providing additional training if needed.

2. Periodic Training

Periodic training should include a description of the content and scheduling of training sessions given for the purpose of ensuring (1) the knowledge and proficiency of radiographers and radiographers' assistants with respect to new regulations, procedures, policies, and equipment and (2) continuing proficiency with present equipment and procedures. Periodic training should be conducted at least annually.

3. Testing Procedures

A description of each test to be given should be submitted. Radiographers must successfully complete a written test and a field examination of the subjects covered in training. Radiographers' assistants must successfully complete a written or oral test and a field examination of the subjects covered in training. A description of an oral or written test may be given by submitting a sample test with an answer for each question.

The effectiveness of any test is reduced if the test is given repeatedly so that the students gain knowledge of its content. The course description should clearly indicate that each test is a sample only and that the test content will be changed at a stated minimum frequency.

A description of an oral examination should be given in the same form as a written examination.

A list of subject areas for the field examination should be given and should include, at a minimum, performance of radiation surveys, posting, operation of equipment, emergency procedures, and other items of radiation safety that may be encountered in the discharge of duties.

A description of the testing procedure is not complete without a clear and specific description of the criteria and procedure to be followed in evaluating test results and determining whether an individual is qualified to act as a radiographer or radiographer's assistant. The relative importance assigned to each question or area of performance, the minimum acceptable number of correct answers or proper responses, and retesting procedures should all be given.

The points within the training program at which each test will be given should be clearly indicated. For example, some training programs include a written or oral exam following the initial training phase, followed by an oral and practical exam at the close of the on-the-job training phase.

Note that testing procedures must ensure compliance with § 34.31. If a trainee is to be utilized as a radiographer's assistant during the training period, the trainee should be tested and qualified as a radiographer's assistant before assuming those duties.

4. Instructor's Qualifications

The person who makes the final determination of the adequacy of a trainee's knowledge and competency should be a qualified radiographer with a strong background of training and experience with radiation. On-the-job training should be given by someone who is, as a minimum, a qualified radiographer.

5. Records

A copy of tests given to each trainee, records showing trainee performance in each examination (including oral and practical examinations), and the examiner's overall evaluation of the trainee as qualified to act as a radiographer or radiographer's assistant must be maintained for a period of three years.

APPENDIX D

PROGRAM RESPONSIBILITIES AND MANAGER QUALIFICATIONS

The individual assigned the duties of maintaining active management control of the program should be a qualified radiographer with training in the use of the types of equipment proposed in the application and should bear the title of Radiation Safety Officer, Radiation Protection Officer, or some similar designation. Thorough knowledge of management policies, company administrative and operating procedures, and safety procedures related to protection against radiation exposures should be prerequisites for the position. Alternates meeting the same qualifications should be designated to assume these functions when necessary.

A list of the duties that may be performed by the licensee's management personnel is presented below. It is not intended to be all inclusive nor should it be interpreted as a requirement that any one person assume all of the listed duties. Some duties may be delegated to persons of lesser authority.

1. Serving as the licensee's liaison officer with the Nuclear Regulatory Commission on license matters.
2. Maintaining control of procurement and disposal of licensed material.
3. Developing and maintaining up-to-date operating and emergency procedures.
4. Establishing and maintaining a personnel monitoring program.
5. Procuring and maintaining radiation survey instruments.
6. Establishing and conducting the training program for radiographers and radiographers' assistants.
7. Examining and determining competence of radiographic personnel.
8. Establishing and maintaining storage facilities.
9. Maintaining exposure devices, radiography facilities, and associated equipment.
10. Establishing and maintaining the leak-testing program.
11. Establishing and maintaining the internal inspection system.
12. Performing source replacement and source tagging operations.
13. Conducting quarterly inventories and maintaining utilization logs.
14. Establishing and conducting a survey instrument calibration program.
15. Establishing and maintaining the licensee's recordkeeping system.

16. Reviewing and ensuring maintenance of those records kept by others.

17. Assuming control and instituting corrective action in emergency situations.

18. Investigating the cause of incidents and determining necessary preventive action.

19. Acting in an advisory capacity to the licensee's management and radiography personnel.

20. Establishing a procedure for evaluating and reporting defects and noncompliance pursuant to 10 CFR Part 21.

EXHIBIT A

Form NRC-313R (7-77) 10 CFR 34	U.S. NUCLEAR REGULATORY COMMISSION **APPLICATION FOR BYPRODUCT MATERIAL LICENSE— USE OF SEALED SOURCES IN RADIOGRAPHY**	Approved by GAO B-180255(R0335)

(SEE ATTACHED FORM NRC-313R INSTRUCTIONS AND NRC REGULATORY GUIDE 10.6—USE SUPPLEMENTAL SHEET WHERE NECESSARY) BE SURE ALL ITEMS ARE COMPLETED AND THAT ALL NECESSARY ATTACHMENTS ARE FURNISHED. IF ANY PORTION OF THE APPLICATION IS NOT APPLICABLE SPECIFICALLY SO STATE. DEFICIENT OR INCOMPLETE APPLICATIONS MAY BE RETURNED WITHOUT CONSIDERATION. LICENSE FEE REQUIRED, SEE ITEM 7 OF INSTRUCTIONS.

1(a) NAME AND ADDRESS OF APPLICANT AND TELEPHONE NUMBER

1(b) TELEPHONE NO.: Area Code (   ) ______ ________

1(c) APPLICANT IS: An individual ☐ A partnership ☐ A Corporation ☐ An Unincorporated Association ☐ Other ☐ If applicant is other than an individual, the applicable section on the reverse side must be completed.

2. THIS IS AN APPLICATION FOR: *(Check appropriate item)*

A. ☐ NEW LICENSE
B. ☐ AMENDMENT TO LICENSE NO. ________
C. ☐ RENEWAL OF LICENSE NO. ________

3. LOCATION(S) WHERE SEALED SOURCES WILL BE USED AND/OR STORED. *(If use will be made in states other than named in 1(a), they should be listed here.)*

4. SEALED SOURCES TO BE USED IN RADIOGRAPHY (Attach supplementary pages, if necessary)

BYPRODUCT MATERIAL *(Element and Mass No.)*	SOURCE MODEL NUMBER	NAME OF MANUFACTURER	MAXIMUM ACTIVITY PER SOURCE	NUMBER OF SOURCES
A.	A.	A.	A.	A.
B.	B.	B.	B.	B.
C.	C.	C.	C.	C.

5(a) RADIOGRAPHIC EXPOSURE DEVICES (Attach supplementary pages, if necessary)

MODEL NUMBER	NAME OF MANUFACTURER *(Include description if custom made)*
A.	A.
B.	B.
C.	C.

5(b) RADIOGRAPHIC SOURCE CHANGERS (Attach supplementary pages, if necessary)

MODEL NUMBER	NAME OF MANUFACTURER *(Include description if custom made)*
A.	A.
B.	B.
C.	C.

6. THE FOLLOWING INFORMATION IS ATTACHED AS A PART OF THIS APPLICATION: *(Check appropriate blocks and attach information called for in the instructions with this form.)*

	Not Applicable	*Attached*	*Previously Submitted*
(a) Description of radiographic facilities (Instruction 6–a)	☐	☐	☐ on ________ (DATE)
(b) Description of radiation detection instruments to be used (Instruction 6–b)	☐	☐	☐ on ________ (DATE)
(c) Instrument calibration procedures (Instruction 6–c)	☐	☐	☐ on ________ (DATE)
(d) Personnel monitoring equipment (Instruction 6–d)	☐	☐	☐ on ________ (DATE)
(e) Operating and emergency procedures (Instruction 6–e)	☐	☐	☐ on ________ (DATE)
(f) Training program (Instruction 6–f)	☐	☐	☐ on ________ (DATE)
(g) Internal inspection system or other management control (Instruction 6–g)	☐	☐	☐ on ________ (DATE)
(h) Overall organizational structure (Instruction 6–h)	☐	☐	☐ on ________ (DATE)
(i) Leak testing procedures (Instruction 6–i)	☐	☐	☐ on ________ (DATE)

CERTIFICATE (This item must be completed by applicant)

7. THE APPLICANT AND ANY OFFICIAL EXECUTING THIS CERTIFICATE ON BEHALF OF THE APPLICANT NAMED IN ITEM 1, CERTIFY THAT THIS APPLICATION IS PREPARED IN CONFORMITY WITH TITLE 10, CODE OF FEDERAL REGULATIONS, PART 30, AND THAT ALL INFORMATION CONTAINED HEREIN, INCLUDING ANY SUPPLEMENTS ATTACHED HERETO, IS TRUE AND CORRECT TO THE BEST OF OUR KNOWLEDGE AND BELIEF.

LICENSE FEE ENCLOSED $ ________

BY: ________
(Signature)

____
(Type or print name of certifying official)

DATE ________

____
(Title of certifying official)

WARNING.—18 U.S.C., Section 1001, Act of June 25, 1948; 62 Stat. 749; makes it a criminal offense to make a willfully false statement or representation to any department or agency of the United States as to any matter within its jurisdiction.

EXHIBIT A page 2

FORM NRC-313R (7-77)

LEGAL STRUCTURE OF APPLICANT

If applicant is a corporation, complete Items 8 through 11; if applicant is a partnership, complete Items 12 through 14; if applicant is an unincorporated association or a legal entity other than a partnership or corporation, complete Items 15 and 16. Attach separate sheets where space provided proves inadequate.

CORPORATION

8. STOCK OF APPLICANT CORPORATION

NO. OF SHARES AUTHORIZED	NO. OF SHARES ISSUED	NO. OF SHARES SUBSCRIBED	TOTAL NUMBER OF:	
			(a) Stockholders	(b) Subscribers

9. Is applicant corporation directly or indirectly controlled by another corporation or other legal entity? YES ☐ NO ☐

If answer is "YES" give name and address of other corporation or other legal entity and describe how such control exists and the extent thereof.

10. (a) Identify by name and address any individual, corporation, or other legal entity (1) owning 10 percent or more of the stock of applicant corporation issued and outstanding or (2) subscribing to 10 percent or more of the authorized but unissued stock of the corporation.

(b) Identify by name and address all officers and directors of the corporation.

11. Identify the State, District, Territory, or possession under the laws of which the applicant is incorporated.

PARTNERSHIP

12. Name and address of each individual or legal entity owning a partnership interest in the applicant.

13. State the percent of ownership of the applicant partnership held by each of the individuals or legal entities listed in Item 12.

14. Identify the State, District, Territory, or possession under the laws of which the applicant partnership is organized.

OTHER

15. Describe the nature of the applicant and identify the State, District, Territory, or possession under the laws of which it is organized.

16. State the total number of members or persons holding an ownership in the applicant, identify each by name and address, and indicate the ownership interest thereof.

DRAFT VALUE/IMPACT STATEMENT

1. BACKGROUND

Industrial radiography accounts for more than two-thirds of overexposures to the whole body greater than 5 rems and to extremities greater than 75 rems and 90% of overexposures to the whole body greater than 25 rems and to extremities greater than 375 rems. While no life-threatening overexposures have been reported, six cases of serious radiation burns to hands have been noted. The general underlying cause of all these exposures was operator failure to make the proper radiation survey to determine source position at the conclusion of a radiographic exposure.

Effective March 3, 1980, 10 CFR Part 34 was revised to reduce operator exposure. The new amendments place stricter requirements on training, supervision, and periodic performance evaluation of radiographers and radiographers' assistants. Also, stricter inspection and survey procedures for exposure devices and sources before use each day and each time the source is retracted to its shielded position will reduce exposures caused by equipment malfunction and operator error. Further requirements on alarms for permanent radiographic installations, on availability of survey instruments, and on use of personnel monitoring devices are designed to make the operator more aware of the hazards of radiation.

2. THE PROPOSED ACTION

2.1 Description

The applicant for a license to use industrial radiography sources is required to develop a program that complies with Commission regulations and to describe this program in the application for the license. Regulatory Guide 10.6 details radiography program requirements and provides guidance in establishing a comprehensive radiation safety program. The proposed revision incorporates the latest amendments to the regulations.

2.2 Need

The recent amendments to 10 CFR Part 34 necessitate extensive revision to the regulatory guide. The amendments and guide changes are as follows:

1. Section 34.11 was amended to require a quarterly internal inspection program of a licensee's operations. The present guide only recommends a program of inspection; revision is necessary to reflect the new requirement.

2. Section 34.22 was amended to require securing of the source in each radiographic exposure device each time the source is returned to its shielded position. The present guide only recommends that such procedures be implemented; revision is necessary to reflect the new requirement.

3. Section 34.28 was amended to require that radiographic exposure devices, storage containers and source changers be checked for obvious damage each day before use and be comprehensively inspected and maintained each quarter. The present guide only recommends that such procedures be implemented; revision is necessary to reflect the new requirement.

4. Section 34.29 was amended to require an alarm system for warning anyone entering a permanent radiographic installation during source exposure. The present guide asks for a description of such a system only if it is present in the facility. Revision of the guide is necessary to reflect the new requirement.

5. Section 34.31 was amended to require documented testing of radiographers (written and practical) and radiographers' assistants (written or oral and practical). The present guide only recommends testing and does not specify the type; revision is necessary to reflect the new requirement and to specify the manner of testing and the materials to be covered.

6. Section 34.43 was amended to require surveys of radiographic exposure devices, including source guide tube, after each exposure. The present guide only recommends that the survey of the guide tube be made; revision is necessary to reflect the new requirement.

7. Section 34.44 was amended to require the physical presence of a radiographer at the site where sources are being used by radiographers' assistants. The present guide does not address itself to this situation; revision is necessary to reflect the requirement.

2.3 Value/Impact

2.3.1 NRC

The review and approval of applications for use of radiography sources are considerably facilitated by the use of a standard format, the instructions for which are provided in the regulatory guide. The guide clearly details the regulations to be followed and the information required for licensing and implementing an acceptable program for the use of industrial radiography sources. Staff review time is shortened because of the use of the standard format and the reduction of unnecessary correspondence resulting from the lack of sufficient detail in license applications. Revision of the guide will reduce additional correspondence that would result because of the regulation changes.

2.3.2 Other Government Agencies

Other government agencies should not be affected.

2.3.3 Industry

The regulatory guide contributes to the reduction in time required for industry's preparation of a license application. Industry will spend less time trying to interpret NRC regulations and requirements for submission of information. More importantly, the revised guide will provide information for the design and implementation of a more effective radiation safety program, thereby minimizing the radiation exposure to workers.

2.3.4 Public

No impact on the public is foreseen.

2.3.5 Worker

The worker will benefit from the revision of the guide through reduced radiation exposure discussed in Item 2.3.3 above.

2.4 Decision

The revision of this regulatory guide should be initiated because of the benefits previously discussed.

3. TECHNICAL APPROACH

This section is not applicable since the proposed action is procedural, i.e., revision of a regulatory guide necessitated by regulation changes.

4. PROCEDURAL APPROACH

4.1 Alternatives

The regulatory guide presently exists. Revision of the guide is necessary because of amendments to the regulation. The only alternative is to discontinue use of the guide altogether in favor of individual letters to licensees.

4.2 Discussion

A guide is the more effective way to transmit information about regulations and licensing requirements. A guide ensures uniform transmission of information to the licensee. Individual letters would be inefficient and, depending on the reviewing official, may not uniformly convey the same information to each licensee. Continuance and revision of the guide is the most effective alternative.

5. STATUTORY CONSIDERATIONS

5.1 NRC Authority

This guide interprets regulations promulgated in 10 CFR Part 34.

5.2 Need for NEPA Assessment

The proposed action is not a major action as defined by 10 CFR 51.5 and does not require an environmental impact statement.

6. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICIES

No conflicts or overlaps appear to exist.

7. SUMMARY AND CONCLUSIONS

The proposed revision to Regulatory Guide 10.6, when disseminated, will assist the NRC in its review of applications for use of industrial radiographic sources and will provide industry with guidelines for submitting applications and implementing optimum radiation safety programs. The proposed revision should be issued.

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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OFFICE OF STANDARDS DEVELOPMENT
DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

July 1980
Division 1
Task RS 110-5

Contact: J. S. Wiebe (301) 443-5913

NUCLEAR POWER PLANT SIMULATORS FOR USE IN OPERATOR TRAINING

A. INTRODUCTION

Appendix A, "Requalification Programs for Licensed Operators of Production and Utilization Facilities," to 10 CFR Part 55 "Operators' Licenses," establishes overall requirements for the use of simulators used for operator training. Included are requirements specifying the similarity that must exist between a simulator and the facility that the operators are being trained to operate. This regulatory guide describes a method acceptable to the NRC staff for specifying the functional requirements of the simulator and for specifying similarity requirements between the simulator and its reference plant.

B. DISCUSSION

Subcommittee ANS-3, Reactor Operations, of the American Nuclear Society Standards Committee developed a standard containing criteria for the characteristics of simulators used for training operators of nuclear power plants. The standard was approved by the American National Standards Institute (ANSI) Committee N18, Design Criteria for Nuclear Power Plants, and was subsequently approved by the ANSI Board of Standards Review and designated ANSI/ANS 3.5-1979, "Nuclear Power Plant Simulators for Use in Operator Training."

The need for improvements in operator training in the areas of response to abnormal and emergency situations became apparent as a result of operator errors reported in Licensee Event Reports and especially because of the types of operator errors noted in NUREG-0585, "TMI-2 Lessons Learned Task Force

FOR COMMENT

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review, have not been reviewed by the NRC Regulatory Requirements Review Committee, and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by SEP 12 1980

Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

Final Report."[1] Use of the actual plant for training in operator response to accidents would result in additional challenges to the plant's protective features and is therefore undesirable. Thus, the additional training required to improve operator performance should be performed on simulators. For such training to be meaningful, the simulator must be kept current with changes in the reference plant, lessons learned from operating experience, and recommendations of instructors and operator trainees. Subcommittee ANS-3 has recognized the need to revise ANS 3.5 to address these issues. The urgency of dealing with these issues requires that this regulatory guide be developed in parallel with the effort to revise ANS 3.5. This regulatory guide is being developed to endorse the latest draft revision to ANS 3.5 available, dated March 24, 1980.[2] Changes to the standard will be evaluated, and the regulatory guide will be revised as necessary.

C. REGULATORY POSITION

The criteria for the characteristics of simulators contained in the March 24, 1980, draft of ANSI/ANS 3.5, "Nuclear Power Plant Simulators for Use in Operator Training," provide an adequate basis for specifying functional requirements of the simulator and for comparing the simulator to its reference plant, subject to the following:

1. Although the title specifies "operator training," this regulatory guide should be applied to simulators used not only for the training and requalification of operators and senior operators, but also for the training and requalification of engineers and management when their position or duties require that they obtain an operator or senior operator license.

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[1] Copies are available for public inspection or copying for a fee at the NRC Public Document Room, 1717 H Street NW., Washington, D.C., or copies may be purchased for $2.50 directly from NRC by sending check or money order payable to the Superintendent of Documents, to the Director, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

[2] Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, LaGrange Park, Illinois 60525.

2. The draft standard identifies at the point of reference (Section 1.1, "Background Data") other documents to be included as part of the standard. The applicability of these documents should be determined by referring to the latest revision of the following regulatory guides and the version of the standard the guide endorses:

ANS Standard[2]	Regulatory Guide
3.1	1.8
3.2	1.33

3. Section 3, "General Requirements," states, in part, that "the extent of simulation shall allow the operator to fully participate in appropriate plant evolutions...." This statement should be taken to mean that the simulator is so designed that the operator trainee may take the same actions to effect an evolution on the simulator as on the reference plant, using similar procedures. Taking the prescribed action, improper action, or no action by the operator trainee should result in the same conditions or automatic actions on the simulator as on the reference plant.

4. The list in Section 3.1.1 of the minimum evolutions to be performed by the simulator should be supplemented as follows:

a. The seventh evolution listed is "Power operations with less than full reactor coolant flow." In addition to simulating power operations, reactor startup and reactor shutdown operations should be simulated in a reduced flow situation if permitted by the reference plant license.

b. The eighth evolution, "Plant shutdown and cooldown to cold (refueling) conditions," should be interpreted to mean "plant shutdown from 100% full power to hot standby to cold (refueling) conditions" in order to ensure that the entire evolution is capable of being simulated.

5. Section 3.1.2, "Plant Malfunctions," lists plant malfunctions to be simulated. Specific malfunctions that are to have an adjustable rate and variable severity capability should be specified by the purchaser. Item (3), "Loss of electrical power (and/or degraded power sources)," should be clarified by including partial or complete failures of power supplies (a) to systems and instrumentation important to safety and (b) to systems and instrumentation not important to safety. The failures should include a.c. or d.c. power supplies

or both as applicable to the reference plant's power supply arrangement and anticipated operational occurrences and accidents.

6. Section 3.2.1, "Control Panels," describes the extent of similarity required between the simulator and reference plant control panels. The criteria should be supplemented as follows:

a. Panels and consoles that are simulated should be designed to duplicate the size, shape, color, and configuration of the reference plant.

b. Face-front hardware (hardware located in or on the simulated control room panels to which the operator would have normal access) and its layout should duplicate that in the reference plant control room.

7. Interactions between other systems and those systems required to be included by Section 3.3.1, "Systems Controlled from the Main Control Panels," should be simulated. These system interactions should provide integrated plant simulation in accordance with the performance criteria of Section 4 of the standard.

8. Appendix A, "Procedure for Documenting Simulator Performance," to the standard should be considered an integral part of the standard.

9. The terms transients, abnormal conditions, abnormal operations, or abnormal evolutions used in the standard should be considered equivalent to "anticipated operational occurrences." The terms emergency conditions or emergency evolutions used in the standard should be considered equivalent to "accidents."

D. IMPLEMENTATION

This proposed guide has been released to encourage public participation in its development. Except in those cases in which an applicant proposes an acceptable alternative method, the method to be described in the active guide reflecting public comments will be used in the evaluation of all simulators placed in operation for training nuclear power plant operators and others whose duties require an operator or senior operator license after the implementation date to be specified in the active guide. In addition, all simulators used for training those who require an operator or senior operator license will be evaluated in accordance with Appendix B, "Implementation Schedule," to the standard except that each of the time periods specified will commence with the date of issuance of the active guide (expected to be about July 1981).

DRAFT VALUE/IMPACT STATEMENT

1. THE PROPOSED ACTION

1.1 Description

Prior to issuing an operator's license to an applicant, the Commission regulations require that evidence be shown that the applicant has learned to operate the controls in a competent and safe manner. In accordance with ANSI/ANS 3.1-1978, "Qualification and Training of Personnel for Nuclear Power Plants," and Regulatory Guide 1.8, "Personnel Selection and Training" (Proposed Revision 2 issued for comment February 1979), reactor simulators may be used to partially fulfill this requirement. In addition, Appendix A to 10 CFR Part 55 describes the use of simulators for requalification programs, and NRC report NUREG-0094, "NRC Operator Licensing Guide,"[3] describes the use of simulators for initial licensing. The proposed action would provide guidance on the acceptable characteristics of reactor simulators used for operator training as described in the above references.

1.2 Need for Proposed Action

The need for improvements in operator training in the areas of abnormal and emergency training became apparent as a result of operator errors reported in Licensee Event Reports and especially because of the types of operator errors noted in NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report." Accident training on the actual plant would result in additional challenges to its protective features and is therefore undesirable. Thus, additional training required to improve operator performance should be performed on simulators. A recommendation to require simulator training for initial and requalification training was made to the Commission and accepted. The simulators used for current training and proposed future training should have characteristics that allow the objective of training the operator to operate the controls in a competent and safe manner to be met. The action proposed by this value/impact statement is expected to ensure that the above objective will be met and also to establish an NRC position on the draft revision of ANS 3.5 as was recommended to and accepted by the Commissioners.

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[3] Copies are available for public inspection or copying for a fee at the NRC Public Document Room, 1717 H Street NW., Washington, D.C., or copies may be purchased for $4.00 directly from NRC by sending check or money order payable to the Superintendent of Documents, to the Director, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

1.3 Value/Impact of Proposed Action

1.3.1 NRC

Section 5 of ANSI/ANS 3.1-1978 (endorsed by Regulatory Guide 1.8) "Selection and Training of Nuclear Power Plant Personnel" uses the adjectives "suitable" (Paragraph 5.2.1) or "appropriate" (Paragraphs 5.2.4 and 5.5.1.2.2) when describing reactor simulators used for training personnel. The value of the proposed action to the NRC will be the availability of guidance on what is a "suitable" or "appropriate" reactor simulator against which to review licensee training programs for adequacy. The impact of the proposed action on the NRC will be the time spent in developing the guidance, reviewing the licensee's proposals to comply with the guide, and verifying implementation of the proposals.

1.3.2 Other Government Agencies

The proposed action should not impact other government agencies, unless the government agency is an applicant, such as TVA.

1.3.3 Industry

The result of the proposed action is expected to be the addition of required functions to simulators that may now be in use for specific nuclear power plants. The impact on industry will likely be increased cost as more complex simulators are required. An alternative to simulators is the use of the actual power plant for training. This use results in lost revenue because the power plant is not available to produce electrical power and also results in additional challenges to the plant protection systems. The value of the proposed action to industry should be (1) more efficient operation of the power plant by better trained operators that would result in a cost savings when the power plant is put on line in an expeditious manner and (2) a reduction in operator errors that might cause plant downtime or equipment damage. The value is based on using simulators that more accurately reflect the power plant they represent and that simulate additional accidents, transients, and evolutions in a more complete manner than can safely be performed on an actual plant.

1.3.4 Public

The value to the public will be that better trained operators should reduce the possibility of improper operation of nuclear power plant equipment or systems that could endanger public health and safety. The impact will be slightly higher electrical rates caused by higher costs as explained in Item 1.3.3 above.

1.4 Decision On Proposed Action

Guidance on the acceptable characteristics of reactor simulators should be provided.

2. TECHNICAL APPROACH

The decision to use simulators for training of operators has been made by the Commission.

3. PROCEDURAL APPROACH

3.1 Procedural Alternatives

1. Regulation
2. Regulatory Guide
3. ANSI Standard, endorsed by a Regulatory Guide

3.2 Value/Impact of Procedural Alternatives

The value of alternative (1), a regulation, is that it would have the full force and authority of a law. The impact of alternative (1) is that it would be difficult to obtain approval of a regulation and that a regulation would lack flexibility in implementation. The value of alternative (2), a regulatory guide, is that it would achieve the desired result with suitable flexibility for innovation by licensees. The impact is that it may not take full advantage of the work performed by industry in the area the guide addresses, which may result in a longer development period prior to issuing the guide. The value of alternative (3), a standard endorsed by a regulatory guide, is that it would achieve the desired result taking advantage of the work performed by industry on the draft revision to ANS 3.5, "Nuclear Power Plant Simulators for Use in Operator Training." The impact of alternative (3) is that effort would have to be expended by the NRC in preparing, reviewing, and issuing the regulatory guide. It is estimated, however, that the effort spent on the proposal would be greater if alternatives (1) or (2) were chosen.

3.3 Discussion on Procedural Approach

The proposed action should be accomplished by publishing a regulatory guide endorsing the approved draft revision to ANS 3.5.

4. STATUTORY CONSIDERATIONS

4.1 NRC Authority

Authority for the proposed action would be derived from the safety requirements of the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974 through the Commission's regulations; in particular, Part 55 of 10 CFR applies.

4.2 Need for NEPA Assessment

An environmental impact statement is not required since the proposed action is not a major action that may significantly affect the quality of the human environment.

5. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICIES

Regulatory Guide 1.8 and Appendix A to 10 CFR 55 make reference to simulators used in the training program for operators. Only general statements are made concerning the characteristics of acceptable simulators. The proposed action will be consistent with and provide additional guidance in this area.

6. SUMMARY AND CONCLUSION

A regulatory guide providing guidance on the characteristics of nuclear power plant simulators for use in operator training should be prepared. The guide should endorse, with possible exceptions, the draft revision to ANS 3.5.

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1. ANSI/ANS 3.1-1978, "Standard for Qualification and Training of Personnel for Nuclear Power Plants."
2. ANSI N18.7-1976/ANS 3.2, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants."
3. ANSI/ANS 3.5-1979, "Nuclear Power Plant Simulators for Use in Operator Training."
4. Draft revision to ANS-3.5, "Nuclear Power Plant Simulators for Use in Operator Training," dated March 24, 1980.
5. Memorandum from S. J. Chilk to L. V. Gossick about SECY 79-330E dated November 29, 1979.
6. NUREG-0094, "NRC Operator Licensing Guide."
7. NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report."
8. Regulatory Guide 1.8, "Personnel Selection and Training."
9. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."
10. SECY 79-330E, "Qualifications of Reactor Operators," dated July 30, 1979, NRC Public Document Room, 1717 H Street NW., Washington, D.C.
11. Title 10 Code of Federal Regulations Part 55, revised January 1, 1979, U.S. Government Printing Office, Washington, D.C. 20402.

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DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

July 1980
Division 3
Task FP 818-4

Contact: D.F. Harmon (301)443-5910

STANDARD FORMAT AND CONTENT OF LICENSE APPLICATIONS, INCLUDING ENVIRONMENTAL REPORTS, FOR IN SITU URANIUM SOLUTION EXTRACTION

FOR COMMENT

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by SEP 30 1980

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INTRODUCTION

A Nuclear Regulatory Commission (NRC) source material license is required under the provisions of Title 10, Code of Federal Regulations, Part 40, "Domestic Licensing of Source Material," to recover uranium by in situ solution extraction techniques (in situ leaching). An applicant for a commercial-scale license or for the renewal or amendment of an existing license to receive, possess, and use source material is required to provide detailed information on the facilities, equipment, and procedures to be used and an environmental report that discusses the operation's impact on the health and safety of the public and on the environment. This information is used by the Commission to determine whether the applicant's proposed activities will, among other things, result in undue risk to the health and safety of the public or adversely affect the environment. General guidance for filing an application and an environmental report is provided in §40.31, "Applications for Specific Licenses," of 10 CFR Part 40 and in 10 CFR Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection," respectively. The purpose of this guide is to provide specific guidance on the format and content of an application, including an environmental report, for a commercial-scale in situ uranium solution extraction facility license. License applications for the research and development study of in situ uranium solution extraction are not addressed in this guide. Applications for licenses authorizing such studies are treated in a similar but less comprehensive manner since such activities are not considered to be major Federal actions.

This guide is intended to provide instructive guidance. It should not be considered as a substitute for a careful evaluation of a program proposed by an applicant.

Information not specifically discussed in this guide should be included in the application if it is a part of an applicant's proposed or existing health and safety or environmental protection program. An incomplete application will result in processing delay and may result in the rejection of a license application.

Changes to existing licensed activities and conditions require the issuance of an appropriate license amendment. An application for such an amendment should describe the proposed changes in detail, discuss the potential environmental and health and safety impacts, and conform to this Standard Format.

Filing an Application

The National Environmental Policy Act of 1969 (83 Stat. 852), implemented by Executive Order 11514 and the Council on Environmental Quality regulations of July 30, 1979 (44 FR 55978), requires all agencies of the Federal Government to prepare detailed environmental statements on proposals for legislation and other major Federal actions significantly affecting the quality of the human environment. The principal objective of the National Environmental Policy Act of 1969 is to build into the agency decisionmaking process an appropriate and careful consideration of the environmental impacts of the proposed actions. 10 CFR Part 51 sets forth the NRC licensing and regulatory policies and procedures for the preparation and processing of environmental impact statements and related documents in accordance with the National Environmental Policy Act.

The provisions of paragraph 40.31(f) of 10 CFR Part 40 and of 10 CFR Part 51 require the submittal of both a license application (Form NRC-2) and a separate environmental report for activities requiring an NRC source material license, which include commercial-scale in situ uranium solution extraction operations. In view of the nature of an in situ uranium solution extraction operation, where a major consideration of both an applicant's submittal and the staff's review is the assessment of environmental impacts of the proposed activity, it appears reasonable that an application and environmental report for a commercial-scale in situ uranium solution extraction license should consist of a single document (hereinafter referred to as the application) containing the information discussed herein.

An application for a new commercial-scale license should be filed at least 12 months prior to planned construction for the proposed operation. An application for a renewal of an existing license should be filed at least 30 days prior to the expiration date of the existing license. An application for an amendment to an existing license should be filed with sufficient lead time to permit a detailed assessment by the NRC staff and issuance of the required authorization before the proposed modification is scheduled to be implemented. All applications must be accompanied by a remittance in the full amount of the fee specified in §170.31 of 10 CFR Part 170, "Fees for Facilities and Materials Licenses and Other Regulatory Services Under the Atomic Energy Act of 1954, As Amended." Applications may be filed with the Director, Office of Nuclear Materials Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 or may be filed in person at the Commission's Offices at 1717 H Street NW., Washington, D.C., or 7915 Eastern Avenue, Silver Spring, Maryland.

Section 51.40 of 10 CFR Part 51 requires an applicant for a license authorizing in situ uranium solution extraction to submit to the Director of Nuclear Materials Safety and Safeguards 15 copies of the application described above. The applicant is also required to retain an additional 85 copies of the application for distribution to Federal, State, and local authorities in accordance with written instructions issued by the Director of Nuclear Materials Safety and Safeguards.

Commercial-scale in situ uranium solution extraction licenses are generally issued for 5-year periods and are renewable over the life of the project. License renewal applications are processed in a manner similar to that used for new applications. Operational experience, site-specific-generated data, and proposed continuing activities are the primary factors considered by the NRC staff in processing renewal applications.

Presentation of Information

The applicant should strive for clear, concise presentation of the information in the license application. Each subject should be treated in sufficient depth and with sufficient documentation* to permit the Commission to independently evaluate the information presented. Tables, line drawings, and photographs should be used wherever they contribute to the clarity and brevity of the application. The number of significant figures stated in numerical data should reflect the accuracy of the data. Descriptive and narrative passages should be brief and concise. In cases where test results to support conclusions are presented, the procedures, techniques, and equipment used to obtain the test data should be included.

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*Documentation as used in this guide means presentation of information, supporting data, and statements and includes (1) references to published information, (2) citations from the applicant's experience, and (3) references to unpublished information developed by the applicant or the applicant's consultants. Statements not supported by documentation may be acceptable provided the applicant identifies them as such or as expressions of belief or judgment.

Information previously submitted to the Commission may be incorporated into the application by reference. Each reference should be clear and specific, i.e., the reference should indicate by document, date, page, and paragraph the information the applicant wishes to reference and how such information is pertinent.

Pertinent published information relating to a proposed site or facility and its surroundings should be referenced. Where published information or assumptions may be essential to evaluate specific aspects of the proposed activities, they should be included in summary or verbatim form or as an appendix to the application.

An in situ uranium solution extraction operation may also be the site for other licensed activities. A separate license will be required for a commercial-scale in situ uranium solution extraction project at mill sites. Amendment of an existing license to include an in situ solution extraction activity will not be considered. In preparing an application relating to in situ uranium solution extraction, the applicant should consider the cumulative or synergistic effects of directly associated activities.

All pages of the application should be numbered and dated.

Contents of an Application

The application should contain the information specified in items 1 through 8 of Form NRC-2. The information required in items 9 through 14 of Form NRC-2 should be incorporated into the various items identified in the chapters of this Standard Format that primarily address processing and environmental considerations. The following environmental concerns must also be fully addressed in these chapters:

1. The environmental impact of the proposed action,
2. Any adverse environmental effects that cannot be avoided if the proposal were implemented,
3. Alternatives to the proposed action,

4. The relationship between local short-term uses of man's environment and the maintenance and enhancement of long-term productivity, and

5. Irreversible and irretrievable commitments of resources involved in the proposed action if it were implemented.

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1. PROPOSED ACTIVITIES

This chapter of the application should summarize the overall proposed activities for which a license is requested in sufficient detail to permit the reviewer to obtain a basic understanding of the proposed activities and potential environmental impact. Review of the chapters that follow can then be accomplished with better perspective and with recognition of their relative importance to the overall operations. The following subjects should be addressed: the corporate entities involved; the location of the proposed activities; ore-body locations and U_3O_8 content; proposed solution extraction method and recovery process; operating plans, design throughput, anticipated annual U_3O_8 production; estimated schedules for construction, startup, and duration of operation; plans for project waste management and disposal; plans for ground-water-quality restoration, decommissioning, and land reclamation; and surety arrangements covering eventual facility decommissioning, ground-water-quality restoration, and site reclamation.

2. SITE CHARACTERISTICS

This chapter should present the basic relevant information concerning those physical, biological, human, and social characteristics of the environs that might be affected by the proposed operations. To the extent possible, the information presented should reflect observations and measurements made over a sufficient period of time to allow defensible conclusions to be reached.

2.1 Site Location and Layout

Provide a map showing the site and its location with respect to State, county, and other political subdivisions. On detailed maps, show the location of the proposed in situ uranium solution extraction operations; exclusion area boundaries and fences; applicant's property; adjacent properties, including water bodies, wooded areas, and farms; nearby settlements; and transportation links (railroads, highways, waterways). Indicate total acreage owned or leased by the applicant and that part occupied or modified by the proposed activity. Also indicate other existing and proposed uses of applicant's property and the acreage devoted to these uses. A contour map of the site should be supplied with elevation contours of an interval suitable to show significant variations of the site environs and drainage gradients. For clarity, this information should be supplied on separate maps.

2.2 Uses of Adjacent Lands and Waters

Indicate, within an 8-km (5-mi) radius, the nature and extent of present and projected land use (e.g., agriculture, sanctuaries, hunting, grazing, industry, recreation, roads) and any recent trends of changes in population or industrial patterns. Identify any other nuclear fuel cycle facilities located or proposed within an 80-km (50-mi) radius of the site.

Provide in tabular form for each of the 22-1/2-degree sectors centered on each of the 16 compass points, i.e., north, north-northeast, etc., the distances (to a distance of 8 km (5 mi)) from the center of the site to the following:

1. Nearest residence.
2. Nearest site boundary.

Identify the location, nature, and amounts of present and projected surface and ground-water use (e.g., water supplies, irrigation, reservoirs, recreation, and transportation) within 8 km (5 mi) of the site and the present and projected population associated with each use point, where appropriate.

Data on both present and projected future water use(s) should be summarized and tabulated; users should be located on maps of legible scale. Tabulations should include:

1. Location: Include symbols shown on maps identifying the location of water users. Provide map coordinates if appropriate.

2. Distances from proposed uranium solution extraction well fields.

3. Withdrawal Rate: Provide present and projected withdrawal rate for each water use. For ground-water uses, indicate depth of wells, ground-water elevations, flow rates, flow directions, and drawdown, and characterize the uses by aquifers.

4. Return Rates: Provide present and projected return rates if appropriate.

5. Type of Water Use: Provide the type of water use for each location, e.g., municipal, industrial, irrigation, stock and game watering.

6. Source and Projection of Water-Use Estimates: Where use rates are anticipated to change over the life of the project and beyond, indicate projections and the source of the projection information. Sources for such projections may be available from users or planning agencies at different levels of government.

7. Abandoned Wells: Furnish a tabulation of all abandoned wells and drill holes giving the location, depth, type of use, condition of closing, plugging procedure used, and date of completion for each well or drill hole within the site area and within 8 km (5 mi) of the site to the extent that such information is available in public records and from reasonable inspection of the area.

For items 3 and 4 above, if seasonal use varies significantly, indicate monthly values.

2.3 Population Distribution

Population data presented should be based on the most recent census data. On a map of suitable scale that identifies places of significant population grouping such as cities and towns within an 80-km (50-mi) radius, concentric circles should be drawn, with the site at the center point, at distances of 1, 2, 3, 4, 5, 10, 20, 30, 50, 70, and 80 km. The circles should be divided into 22-1/2-degree sectors with each sector centered on one of the 16 compass points, i.e., north, north-northeast, northeast, etc. A table appropriately keyed to the map should provide the current residential population within each area and for census years through the anticipated life of the operation. The table should provide separate and cumulative population totals for each sector and annular ring. Distance to the nearest residence should be noted for each sector. The basis for population projections should be described.

Descriptive material should include tables giving significant population and visitor statistics of neighboring schools, plants, hospitals, sports facilities, residential areas, parks, etc., within 8 km (5 mi) of the solution extraction operations. The material should also include appropriate available food production data in kg/yr for vegetables (by type and totals), meat (all types), and milk and any available future predictions by local governmental, industrial, or institutional organizations.

2.4 Regional Historic, Archeological, Architectural, Scenic, Cultural, and Natural Landmarks

The application should include a brief discussion of the historic, scenic, archeological, architectural, cultural, and natural significance, if any, of the proposed site and nearby areas with specific attention to the site and nearby areas listed in the National Registry of Natural Landmarks and properties included in or eligible for inclusion in the National Register of Historic Places.

The National Registry of Natural Landmarks appears in the Federal Register (37 FR 1496). The National Register of Historic Places is published annually in the Federal Register; additions are published in the Federal Register on the first Tuesday of each month. General guidance on the treatment of historic, archeological, architectural, and cultural features can be obtained from the National Park Service publication entitled "Preparation of Environmental Statements: Guidelines for Discussion of Cultural (Historic, Archeological, Architectural) Resources," August 1973.*

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*Copies may be obtained from Chief Historian, Room 1226, National Park Service, 18th and C Streets NW., Washington, D.C. 20240.

The application should identify those properties included in or eligible for inclusion in the National Register of Historic Places located within the area of the proposed project. The applicant should also consult with the appropriate State Historic Preservation Officer (SHPO) concerning the identification of properties included in or eligible for inclusion in the National Register of Historic Places. The application should contain evidence of contact with the Historic Preservation Officer for the State involved. A copy of the SHPO's comments concerning the effect of the facility on historic, archeological, architectural, and cultural resources should be included in the application.

State whether new roads, pipelines, and utilities for the proposed activity will pass through or near any area or location of known historic, scenic, cultural, natural, archeological, or architectural significance.

2.5 Meteorology

This section should provide a description of the meteorological diffusion characteristics of the site and its surrounding area. The description should be based on data collected on site or at nearby local meteorological stations. Sufficient data should be included to permit independent evaluations and assessments of atmospheric diffusion characteristics.

The following data concerning site meteorology from meteorological measurements taken on site and at nearby representative stations should be provided:

1. Joint frequency data
    a. National Weather Service (NWS) station data
        (1) Locations of all NWS stations within an 80-km (50-mi) radius
        (2) Available joint frequency distribution data by wind direction, wind speed, and stability class (3-dimensional numerical array)
        (3) Period of record by month and year
        (4) Height of data measurement
    b. Onsite meteorological data
        (1) Locations and heights of instrumentation
        (2) Description of instrumentation
        (3) Minimum of 1 full year of onsite joint frequency distribution data broken down by wind direction, wind speed, and stability

class (3 dimensional array) with a joint data recovery of 90 percent or more

2. Miscellaneous data
    a. Annual average mixing depth heights
    b. Description (general) of regional climatology, particularly including frequencies and durations of extreme wind speeds
3. Total precipitation and evaporation by month

This information should be fully documented and substantiated as being representative of expected long-term conditions at and near the site.

The joint wind speed-stability-direction frequencies should be presented in tabular form, giving the frequencies as fractions when using 5-year NWS summaries or as the number of occurrences when using only 1 or 2 years of onsite data. The data should be presented for each of the 16 compass directions, and the stability categories should be established to conform as closely as possible with those of Pasquill. In addition, the annual average inversion height should be provided from other nearby weather stations.

Guidance on acceptable onsite meteorological measurements and data format is presented in Regulatory Guide 1.23, "Onsite Meteorological Programs."

In addition, this section should provide a discussion of general climatology, existing levels of air pollution, the relationship of the meteorological data gathered on a regional basis to local data, the impact of the local terrain and large lakes and other bodies of water on meteorological conditions in the area, and the occurrence of severe weather in the area and its effects. Data on diurnal and monthly averages of temperature and humidity should also be provided.

2.6 Hydrology

The effects of construction and operation on adjacent surface waters and ground waters and the effective control and monitoring of subsurface process fluids are of prime importance. The applicant should describe in quantitative terms the physical, chemical, biological, radiological, and hydrological characteristics, the typical seasonal ranges and averages, and the historical extremes for surface-water bodies and aquifers associated with the proposed project. Water quality data should include measurements made at and in close proximity (within 200-400 feet) of the proposed in situ uranium solution extraction areas (well fields).

2.6.1 Ground Water

The hydrology of both regional and local ground-water systems should be described. The description of the ground-water setting should include identification of the average thickness, lateral extent, general flow direction, average yield, and average potentiometric pressure of the regional or affected aquifers.

Within the local ground-water systems, all aquifers that may be affected by the proposed in situ uranium solution extraction operations should be identified. The hydrologic properties of the local aquifers, including aquifer thickness, distribution of potentiometric levels, flow gradients, flow direction, flow velocity, directional permeabilities, transmissivities, storage coefficients, and porosities should be described in detail. The confining beds between the ore zone(s) and other aquifers, including continuity of vertical permeabilities, horizontal permeabilities, and other data sufficient for evaluation of the confining properties of the beds, should also be discussed. A description of soil types and near-surface material, including hydrologic properties, should be presented in sufficient detail to permit evaluations of the effects of surface activities related to the proposed uranium extraction operations. Conclusions concerning the hydrologic characteristics of site aquifers, confining layers, and soil type should be based on well borings, pump tests, soil surveys, and results of other appropriate investigations.

Descriptions of local ground-water well uses, including well location, depth, screened intervals, yield, static water level, and preoperational water quantity used, should be presented in the application. The descriptions should be in sufficient detail to fully define the use of ground water in the project environs.

The preoperational water quality of all aquifers that might be affected by the proposed operations, as well as the changes expected in quality due to the solution extraction activities, should be described.

Data from pump tests, water quality measurements, and other tests should be furnished to substantiate hydrologic interpretations.

2.6.2 Surface Water

Describe the location, size, shape, and hydrologic characteristics and uses of surface-water bodies in the environs of the site.

Include a description of upstream and downstream river control structures, and provide a topographic map showing the major hydrologic features. Water quality analysis and flow rates from U.S. Geological Survey gaging stations in nearby environs should be obtained and included.

2.7 Geology

A description of the geology of the site and establishment of the continuity of the geologic environs represented in the rocks at the site should be included in the application. The discussion should note local and regional stratigraphy, structure, and tectonic history. Comments should be made as appropriate on dips, faulting, fracturing, and continuity of geologic environments within the site and nearby regions. Structural maps, cross sections, and geologic columns of the site and its environs should be included. Conclusions concerning the geology should be based on well borings, geophysical surveys, remote-sensing measurements, and the results of other appropriate investigations. Geologic logs, geophysical logs, and other data should be furnished in an appendix. Proprietary data should be so designated and kept separate from the remainder of the application.

An inventory of economically important minerals and energy-related deposits, in addition to the uranium ore, should be included in the discussion. Any unique minerals (including those that might be affected by fluid movement associated with the proposed project such as bentonites) or paleontological deposits of particular scientific interest should also be identified. Any effect that planned operations might have on the future availability of other mineral resources should be noted.

2.8 Seismology

Discuss the seismicity (including its history) of the region. Where possible, associate seismic events with tectonic features identified in the geology discussion. Furnish a regional earthquake epicenter map showing site location.

2.9 Ecology

In this section, the applicant should describe the flora and fauna in the vicinity of the site, their habitats, and their distribution. This inventory should identify species that may require specific attention because of their importance to the community. A species is "important" (for the purposes of this guide) if a specific causal link can be identified between the facility and the species and if one or more of the following criteria applies: (1) the species is commercially or recreationally valuable, (2) the species is threatened or endangered,* (3) the species affects the well-being of some important species within criterion (1) or (2), or (4) the species is critical to the structure and function of the ecological system or is a biological indicator of radionuclides or chemical pollutants in the environment.

The inventory should establish the identity of the majority of the terrestrial and aquatic organisms on or near the site and their relative (qualitative) abundance. The applicant should identify the "important" species from this list and discuss in detail their quantitative abundance. This discussion should include species that migrate through the area or use it for breeding grounds. Special attention should be given to the relative importance of the proposed site environs to the total regional area for the living resources (potential or exploited).

The applicant should provide data on the count and distribution of important domestic fauna, in particular, cattle, sheep, and other meat animals that may be involved in the exposure of man to radionuclides. Important game animals should receive similar treatment. A map showing the distribution of the principal plant communities should be provided.

The discussion of species-environment relationships should include descriptions of area usage (e.g., habitat, breeding) for important species; life histories of important regional animals and aquatic organisms, their normal seasonal population fluctuations, and their habitat requirements; and identification of food chains and other interspecies relationships, particularly when these are contributory to predictions or evaluations of the impact of the facility on the regional biota.

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*In the writing and reviewing of environmental reports, specific consideration should be given to possible impact on any species (or its habitat) that has been determined to be endangered or threatened with endangerment by the Secretary of the Interior and the Secretary of Commerce. New terminology defining "endangered or threatened with endangerment" has been issued in Public Law 93-205, 87 Stat. 884.

Any definable preexisting environmental stresses from sources such as pollutants, as well as pertinent ecological conditions suggestive of such stresses, should be identified. The status of ecological succession should be described. Discussion should include the histories of any infestations, epidemics, or catastrophes (caused by natural phenomena) that have had a significant impact on regional biota.

The information should be presented in two separate subsections: "Terrestrial Ecology" and "Aquatic Ecology." The sources of information should be identified. As part of this identification, a list of pertinent published material dealing with the ecology of the region should be presented. All ecological or biological studies of the site or its environs currently in progress or planned should be referenced and described.

2.10 Background Radiological Characteristics

Report regional radiological data, including both natural background radiation levels and results of measurements of concentrations of radioactive materials occurring in important biota, in soil and rocks, in air, and in regional surface water and local ground waters that could be affected by the proposed activities. These data, whether determined during the applicant's preoperational surveillance program or obtained from other sources, should be referenced.

2.11 Background Nonradiological Characteristics

Regional nonradiological characteristics, particularly those that are related to expected site-related effluents, should be reported. Data should include such parameters as heavy metals and other potentially toxic substances in surface and subsurface waters, atmospheric pollutants, dusts, etc., that could affect water or air quality. Other regional sources of these same materials should be noted along with a discussion of the possible incremental contribution to the existing levels found at the site.

2.12 Other Environmental Features

Some relevant information on the environs may not clearly fall within the scope of the preceding topics. Additional information may be required with respect to some environmental features in order to reflect the value of the site and site environs to important segments of the population. Such information should be included here.

3. DESCRIPTION OF PROPOSED FACILITY

The in situ uranium solution extraction operation should be described in this chapter. Since environmental effects are of primary concern, the combined effects of extraction effluents and related systems that interact with the environment should be described in sufficient detail to permit an independent evaluation by the NRC of the proposed project.

3.1 Solution Extraction Process and Equipment

The entire in situ solution extraction process should be described in sufficient detail to permit evaluation of all operations and processes involved. This description would include data about the ore bodies, the feasibility of processing defined well-field areas, procedures to ensure that well installations will not result in hydraulic communication between production zones and overlying or underlying aquifers, how wells and ponds will be completed, injection/production rates and pressures, proposed operating plans and schedules, details of the proposed uranium recovery facility and its operation, plant material balances and flow rates, and major constituents of the gaseous, liquid, and solid wastes and effluents that will be generated in the process. The following should also be provided:

1. A map or maps showing the proposed sequence and schedules for the in situ uranium solution extraction well-field areas and well-field ground-water-quality restoration operations.
2. A flow diagram of the process and/or circuit.
3. A material balance diagram.
4. A description of any chemical recycle systems.
5. A water balance diagram for the entire system.
6. A map or maps showing the proposed sequence and schedules for land reclamation of the well-field areas.

3.2 Recovery Plant Equipment

A physical description and operating characteristics for the proposed major items of process equipment should be provided. A diagram of the proposed plant layout, indicating areas and points where dusts, fumes, or gases would be generated, should be included. The diagram should also show the locations of all ventilation, filtration, confinement, and dust collection systems, as well as the location of the radiation monitoring equipment identified in Chapters 4, "Effluent Control Systems," and 5, "Operations."

3.3 Instrumentation

A description of proposed process instrumentation and control systems relevant to safety and radiation safety sampling and monitoring instrumentation, including their minimum specifications and operating characteristics, should be provided. Sufficient information should be included to permit an evaluation of the interrelationship between instrumentation systems and the operations or processes to be controlled or monitored.

4. EFFLUENT CONTROL SYSTEMS

4.1 Gaseous

Provide a description of all proposed ventilation, filtration, and confinement systems that are to be used during operations to control the release of radioactive materials to the atmosphere. Include an analysis of equipment as designed and operated to prevent radiation exposures to employees and to limit such exposures to as low as is reasonably achievable (ALARA). Also include a physical description of discharge stacks, types and estimated composition and flow rates of atmospheric effluents, and proposed methods for controlling such release levels ALARA.

4.2 Liquids and Solids

To the extent that information is not provided in Section 3.1, provide a realistic estimate of the quantities and composition of all waste residues expected, along with proposed procedures for their management. Where retention systems such as levees, dikes, and ponds are to be used to prevent the release of liquid or solid wastes containing radioactive material to offsite areas, provide the information specified in Regulatory Guide 3.11, "Design, Construction, and Inspection of Embankment Retention Systems for Uranium Mills." Also provide descriptions and design details for all temporary and permanent surface-water diversion facilities. Describe contingency plans to mitigate any environmental impact in the event that leakage occurs from impoundments containing potentially harmful materials.

5. OPERATIONS

Compliance with the statements, representations, and procedures provided in this chapter will normally be made a specific condition of the NRC source material license. Thus, the following should be considered as specific commitments on the part of the applicant for conducting operations and radiological protection programs. In addition, the bases for all programs addressed in this chapter, as well as the demonstration of their adequacy, should be provided.

5.1 Corporate Organization and Administrative Procedures

Provide a detailed description of the applicant's proposed organization, including authority and responsibility of each level of management and supervision with regard to development, review, approval, implementation, and adherence to operating procedures, radiation safety programs, routine and nonroutine maintenance activities, and changes in any of the above.

5.2 Management Control Program

Describe the proposed management control program and administrative procedures to ensure that all activities are conducted in accordance with written operating procedures that will be approved and reviewed at specified frequencies by the radiation safety staff. This program should provide a method for ensuring that any nonroutine work or maintenance activity not covered by an effective operating procedure will be conducted in accordance with a special work permit reviewed and approved by the radiation safety staff.

5.3 Management Audit and Inspection Program

Describe the proposed management audit and internal inspection program, including frequencies and types and scopes of reviews and inspections, action levels, and corrective action measures with regard to the foregoing activities. Also identify by management position the person responsible for each phase of the audit and inspection program.

5.4 Qualifications

Provide a description of the minimum qualifications and experience required for personnel holding positions in the applicant's proposed organization who will be assigned the responsibility for developing, conducting, and administering the radiation safety program. Also provide in an appendix the qualifications of the persons currently proposed for these positions.

5.5 Training

Provide a description of the proposed employee radiological protection training program, including the content of the initial training or indoctrination, testing, on-the-job training, and extent and frequency of retraining. In an appendix, provide a copy of the proposed written radiological safety instructions to be provided to employees. These instructions should include provisions for personal hygiene (including washing), for contamination surveying prior to eating or leaving the operating area, for wearing personnel monitoring devices and respirators, for good housekeeping requirements, for cleaning up spills within the site boundary, and for emergency action in the event of accidents.

5.6 Security

Provide a description of the proposed method for preventing unauthorized entry into the controlled area.

5.7 Radiation Safety Controls and Monitoring

Paragraph 20.1(c) of 10 CFR Part 20 states that "... persons engaged in activities under licenses issued by the Nuclear Regulatory Commission pursuant to the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974 should, in addition to complying with the requirements set forth in this part, make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to unrestricted areas, as low as is reasonably achievable." Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," provides the NRC staff position on this important subject. License applicants should give consideration to the ALARA philosophy, as described in Regulatory Guide 8.10, in the development of plans for work with licensed radioactive materials. The following descriptive material should be provided:

5.7.1 Effluent Control Techniques

Describe the proposed systems and procedures designed to minimize in-plant and environmental emissions at each step of the process where releases might occur. Include descriptions of the contingency plans to be implemented in the event of equipment failures or spills.

5.7.2 External Radiation Exposure Monitoring Program

Describe the proposed methods, instrumentation, and equipment for determining exposures of employees to external radiation during routine and nonroutine operations, maintenance, and cleanup activities. Also describe the type of surveys to be conducted, criteria for determining survey locations, frequency of surveys, action levels, management audits, and corrective action requirements. For personnel monitoring devices such as film badges, indicate the number and category of personnel involved in the program and the sensitivity and range of the devices. For survey instruments, provide instrument sensitivities, ranges, calibration methods (in an appendix), and frequency of calibration.

5.7.3 Airborne Radiation Monitoring Program

Describe the proposed sampling program to determine concentrations of airborne radioactive materials (including radon) during routine and nonroutine operations, maintenance, and cleanup activities. In the description of the sampling program, include:

1. The criteria for determining sampling locations with respect to process operations and personnel occupancy; and
2. The frequency of sampling, type of analyses, sensitivity of overall sampling and analyses, action levels, management audits, corrective action requirements, and instrumentation calibration frequency. Procedures for sample analyses and instrument calibration should be included in an appendix.

5.7.4 Exposure Calculations

Describe the proposed procedure, in conformance with §20.103 of 10 CFR Part 20, to determine the intake of radioactive materials by personnel in work areas where airborne radioactive materials could exist. Include those exposures incurred during nonroutine operations, maintenance, and cleanup activities as well as during routine activities.

5.7.5 Bioassay Program

Describe the proposed bioassay program to confirm the results derived from the programs identified in Sections 5.7.3 and 5.7.4. Indicate the number and category of personnel involved in the program, the types and frequencies of bioassays performed, and action level criteria to be applied to bioassay results.

5.7.6 Contamination Control Program

Describe the proposed occupational radiation survey program to determine that employees (plus their workclothes or coveralls, etc.) entering clean areas (lunchrooms, offices, etc.) or leaving the site are not contaminated with radioactive materials. The description should include proposed housekeeping and cleanup requirements and specifications in process areas to control contamination; frequency of surveys of clean areas; survey methods; and minimum sensitivity, range, and calibration frequency of survey equipment. Provide proposed contamination criteria or action levels for clean areas and for the release of materials, equipment, and workclothes to clean areas or from the site. Procedures for instrument calibration should be included in an appendix.

5.7.7 Airborne Effluent and Environmental Monitoring Program*

Describe in detail the proposed effluent monitoring program, including methods and procedures for measuring concentrations and quantities of airborne radioactive materials released to the environs. The description of the proposed environmental monitoring program should include the technical basis used to determine environmental concentrations.

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* 40 CFR Chapter 1, Part 190, and Regulatory Guide 4.14, "Radiological Effluent and Environmental Monitoring at Uranium Mills," should be reviewed in connection with this section.

For both effluent and environmental monitoring, the frequency of sampling and analysis, the types and sensitivity of analysis, action levels and corrective action requirements, and the minimum number and criteria for locating effluent and environmental monitoring stations should be provided. Proposed locations should be indicated on a layout of the site and surrounding area.

5.7.8 Liquid Effluent and Environmental Monitoring Program*

Describe the liquid effluent monitoring program to determine if process effluents are in hydrologic communication with subterranean or surface-water supplies. This description should include the technical basis, the minimum number of monitoring stations, the criteria for locating sampling stations, the frequency of sampling, and action levels and corrective action requirements. Provide the procedures for sample analyses of natural uranium, radium-226, and thorium-230 in an appendix.

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*40 CFR Chapter 1, Part 190, and Regulatory Guide 4.14, "Radiological Effluent and Environmental Monitoring at Uranium Mills," should be reviewed in connection with this section.

6. GROUND-WATER-QUALITY RESTORATION, SURFACE RECLAMATION, AND PLANT DECOMMISSIONING

The applicant should describe in detail proposed plans for ground-water-quality restoration, surface reclamation, and plant decommissioning. Detailed discussions should be provided for the following:

1. Plans and schedule(s) for ground-water-quality restoration, including

a. An estimate of the quantities and concentrations of those chemic that may persist in leached-out well-field production zones after termination of in situ extraction operations and prior to restoration activities;

b. A description of proposed methods and techniques to be used to achieve ground-water-quality restoration, including identification of in situ chemical reactions that may hinder or enhance restoration. The applicant shoul provide an analysis of the methods and techniques to be used to achieve restoration in terms of fluids to be used during restoration and the hydraulic and geochemical properties of the receiving stratum;

c. A description of the expected postreclamation conditions and quality of restored ground waters, compared with the preoperational land and water quality characteristics;

d. An assessment of the proposed water quality restoration operatic with respect to adversely affecting ground waters outside production zones; anc

e. The procedures to be used for plugging, sealing, capping, and abandoning of all wells associated with the in situ solution extraction operations.

2. Plans and schedule(s) for reclaiming disturbed lands, including;

a. A contour map showing the approximate postreclamation surface contours for affected lands and immediate surrounding area(s);

b. Procedures for the reclamation of any temporary diversion ditche and impoundments;

c. Procedures for reestablishing any surface drainage that may be disrupted by the solution extraction operations;

d. Procedures for mitigating or controlling the effects of any subsidence; and

e. Procedures for ground surface preparation, depth of topsoil replacement and revegetation plans, erosion control and water conservation practices, and proposed postoperational land use.

3. Procedures for removing and disposing of structures used in conjunction with the in situ solution extraction operations, including procedures for managing all toxic and radioactive materials. In the discussion pertaining to the disposal of wastes produced by in situ solution extraction operations, procedures for complying with criterion 1 (i.e., removal and disposal of byproduct material at an existing uranium mill tailings disposal site) of proposed Appendix A to 10 CFR Part 40 (44 FR 50015) should be included.

4. Procedures for conducting postreclamation and decommissioning radiological surveys to ensure that sufficient potential radioactive contamination has been removed from the site to permit its release for unrestricted use. Include plans for postoperational ground-water monitoring, if appropriate, to ensure that restored water quality is stabilized.

5. Financial arrangements to be made to ensure that adequate funds will be available for the ground-water-quality restoration, facility decommissioning, land reclamation, and radiological monitoring described above. Such arrangements should be based on a financial assessment of estimated costs, which should also be described.

7. ENVIRONMENTAL EFFECTS

The construction of facilities and well drilling will inevitably affect the environment; some of the effects may be adverse and others beneficial. Effects are considered adverse if environmental change or stress causes a valuable or otherwise important biotic population or natural resource to be less safe, less healthy, less abundant, less productive, less esthetically or culturally pleasing; if the change or stress reduces the diversity and variety of individual choice, the standard of living, or the extent of sharing of life's amenities; or if the change or stress tends to lower the quality of renewable resources or to impair the recycling of depletable resources.

The applicant's discussion of adverse environmental effects should distinguish between those that are considered unavoidable and subject to later amelioration and those that are regarded as unavoidable and irreversible. Those effects representing an irreversible and irretrievable commitment of resources should receive detailed consideration. (In the context of this discussion, "irretrievable commitment of resources" refers to natural resources and means their permanent impairment, e.g., loss of wildlife habitat; destruction of nesting, breeding, or nursing areas; interference with migratory routes; loss of valuable or esthetically treasured natural areas; and expenditure of directly used resources.)

7.1 Site Preparation and Construction

The applicant should organize the discussion in terms of the effects of site preparation and construction on both land use and water use. The applicant should consider the consequences to both human and wildlife populations and indicate which are unavoidable, irreversible, etc., according to the categorization set forth above.

A description of how construction activities may disturb the existing terrain and wildlife habitats should be included in the land-use discussion. Consider the effects of such activities as building temporary or permanent roads, bridges, or service lines; disposing of trash; excavating; and land filling. Provide information bearing on such questions as how much land will be disturbed, for how long, will there be dust or smoke problems, and what explosives will be used, where, and how often. Indicate proximity of human populations, and identify undesirable impacts on their environment arising from noise, disruption of stock grazing patterns, inconvenience due to the movement of men, material, and machines, including activities associated with any provision of housing, transportation, and educational facilities for workers and their families. Describe any expected changes in accessibility of historic and archeological sites in the region. Discuss measures designed to mitigate or reverse undesirable effects such as erosion control, dust stabilization, landscape restoration, control of truck traffic, and restoration of affected habitats.

The discussion should also include any effects of site preparation and construction activities the consequences of which may be beneficial to the region.

The discussion of water use should describe the impact of site preparation and construction activities on area water sources. The applicant should describe the effects of these activities on fish and wildlife resources, water quality, water supply, esthetics, etc., as applicable. Describe measures to mitigate undesirable effects such as pollution control and other procedures for habitat improvement.

7.2 Effects of Operations

The impacts of operation of the proposed activity should be, to the fullest extent practicable, quantified and systematically presented in this section. In the discussion of each impact, the applicant should make clear whether the supporting evidence is based on theoretical, laboratory, onsite, or field studies undertaken for this or for previous operations. The source of each impact (the plant subsystem, waste effluent) and the population or resource affected should be made clear in each case. The impacts should be distinguished in terms of their effects on surface-water bodies, ground water, air, land, land use, ecological systems, and important plants and animals, etc.

7.3 Radiological Effects

In this section, the applicant should consider the radiological effects of operations on man. Estimates of the radiological impact on man via various exposure pathways should be provided.

7.3.1 Exposure Pathways

The various possible pathways for radiation exposure of man should be identified and described in textual and flow-chart format. Discuss any exposure pathways, if they exist, involving radionuclide accumulation in specific components of the environment.

7.3.2 Exposures from Water Pathways

Estimate the expected annual average concentrations of radioactive nuclides in receiving water at locations where water is consumed or is otherwise used by human beings or where it is inhabited by biota of significance to human food chains. Specify the dilution factors used in preparing the estimates and the locations where the dilution factors are applicable. Consideration should be given to the absence of mixing and dilution because of factors such as channeling.

Estimate the expected radionuclide concentrations in aquatic and terrestrial organisms significant to human food chains. Use bioaccumulation factors as necessary.

Using the above information and any other necessary supporting data, calculate the total annual body and significant organ doses (in millirems) to individuals in the population from all receiving-water-related exposure pathways, i.e., all sources of internal and external exposure. Provide an appendix describing details of the models and assumptions used in these calculations.

7.3.3 Exposures from Air Pathways

From release rates of airborne radioactivity and meteorological data, estimate total annual body and significant organ doses (in millirems) to individuals exposed at the point of maximum ground-level concentrations off site, individuals exposed at the site boundary in the direction of the prevailing wind, individuals exposed at the site boundary nearest to the sources of emission, and individuals exposed at the nearest residence in the direction of prevailing wind. Assume annual average meteorological conditions. Identify locations of points of release (e.g., stack, roof vent) used in calculations.

Estimate deposition of radioactive materials on food crops and pasture grass. Estimate total annual body doses (in millirems) and significant annual doses received by other organs via such potential pathways.

Provide an appendix describing the models used in these calculations.

7.3.4 Exposures from External Radiation

The applicant should provide an estimate of the maximum annual external dose (in millirems) that would be received by an individual from direct radiation at the nearest site boundary.

7.3.5 Total Human Exposures

The applicant should provide estimates of the maximum annual doses (in millirems) that could be received via all pathways described above by an individual at the site boundary and at the nearest residence.

The applicant should also present a table that summarizes the estimated radiation dose to the regional population (within 80 km) from the uranium recovery plant and well-field-related sources using values calculated in previous sections. The tabulation should include (1) the total annual dose (man-rems) to the population from all water-related pathways and (2) the total annual dose (man-rems) to the population attributable to airborne effluents.

7.3.6 Exposures to Flora and Fauna

From considerations of the exposure pathways and the distribution of radioactivity released into the environs, the applicant should estimate the maximum radionuclide concentrations that may be present in important local flora and local and migratory fauna. Values of bioaccumulation factors used in preparing the estimates should be based on site-specific data if available; otherwise, values from the literature may be used. The applicant should tabulate and reference the values of bioaccumulation factors used in the calculations.

7.4 Nonradiological Effects

In this section, the specific concentrations of nonradioactive wastes in effluents at the points of discharge should be compared with natural ambient concentrations without the discharge and should also be compared with applicable standards. The projected effects of the effluents for both acute and chronic exposure of the biota (including any long-term buildup in soils and sediments and in the biota) should be identified and discussed. Dilution and mixing of discharges into the receiving environs should be discussed in detail, and estimates of concentrations at various distances from the point of discharge should be provided. The effects on terrestrial and aquatic environments from chemical wastes that contaminate ground water should be included.

The applicant should also discuss any potential effects of the proposed operation that do not clearly fall under any specific topic delineated above. These may include changes in land and water use at the project site; sanitary and other recovery plant waste systems; interaction of the facility with other existing or projected neighboring facilities; effects of ground-water withdrawal on ground-water resources in the vicinity of the well field(s) and recovery plant(s); effects of construction and operation of roads, transmission corridors, railroads, etc.; effects of changes in surface-water availability on biotic populations; and disposal of solid and liquid wastes other than those already discussed.

7.5 Effects of Accidents

The applicant should discuss the environmental effects of possible accidents that may occur, whether or not those accidents may produce an impact on the site or its environs. Analyses should be based on relevant experience and accident statistics from similar operating facilities. Accidents due to both human causes and natural phenomena should be addressed.

7.5.1 Accidents Involving Radioactivity

The applicant should provide realistic analyses of accidents involving radioactivity for a spectrum of accidents that might occur ranging in severity from trivial (essentially no release of radioactivity to the environment) to large releases. Each class within the spectrum should be characterized by an occurrence rate or probability and its potential consequences, if any. Examples of accidents resulting in large releases would be an undetected lixiviant excursion or the failure of a waste retention system resulting from an act of nature, faulty design, or misoperation. Examples of accidents resulting in small releases would be failure of a pumping circuit with ground surface lixiviant release or failure of the ventilation system serving the chemical makeup area. An example of a trivial accident would be the malfunction of well-head equipment or the leakage of a vessel containing barren lixiviant solution. Also describe measures to be taken to prevent accidents, and provide a discussion of proposed contingency plans to be implemented in the event that accidents occur.

7.5.2 Transportation Accidents

The potential environmental effects from transportation accidents involving radioactive and other hazardous materials should be evaluated.

7.5.3 Other Accidents

In addition to accidents that could release radioactivity to the environs, there might be accidents that, although radioactive materials would not be involved, would have consequences that could affect the environment. Such accidents as chemical explosions or fires, steam boiler failures, and leakage or rupture of vessels containing toxic materials could have significant environmental impacts. These possible accidents and associated effects should be identified and evaluated.

7.6 Economic and Social Effects of Construction and Operation

The purpose of this section is to provide guidance on the information needed to assess the economic and social effects of the proposed operations. There are, of course, limitations on the extent to which the social and economic benefits and costs of a project can be evaluated. The wide variety of benefits and costs are not only difficult to assess, but many are not amenable to quantification or even to estimation in commensurable units. Some primary benefits such as the quantity of uranium recovered are, to a degree, measurable as are the capital costs and operating and maintenance costs of the proposed

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facility. On the other hand, numerous environmental costs and their economic and social consequences are not readily quantified. All potential significant social and economic benefits and costs should be addressed in the application and, to the extent possible, should be discussed in quantitative terms.

7.6.1 Benefits

The primary benefits of the proposed facility are those inherent in the value of the uranium to be recovered and the kilowatt-hours of electricity the uranium represents.

There are other social and economic benefits that affect various political jurisdictions or interests to a greater or lesser degree. Some of these reflect transfer payments or other values that may partially, if not fully, compensate for certain services as well as external or environmental costs, and this fact should be reflected in the designation of the benefit. Some examples are:

- Tax revenues to be received by local, State, and Federal governments.
- Temporary and permanent new jobs created and payroll (value-added concept).
- Incremental increase in regional product.
- Enhancement of recreational values.
- Environmental enhancement in support of the propagation or protection of wildlife and the improvement of wildlife habitats.
- Creation and improvement of local roads, waterways, or other transportation facilities.
- Increased knowledge of the environment as a consequence of ecological research and environmental monitoring activities associated with plant operation and technological improvements from the applicant's research programs.

The applicant should discuss significant benefits that may be realized from construction and operation of the proposed facility. Where the benefits can be expressed in monetary terms, they should be discounted to present worth. In each instance where a particular benefit is discussed, the applicant should indicate, to the extent practical, who is likely to be affected and for how long. In the case of esthetic impacts that are difficult to quantify, the applicant should provide pictorial drawings of structures or environmental modifications visible to the public.

7.6.2 Costs

The economic and social costs resulting from the proposed operations are likewise complex and need to be appraised.

The primary internal costs are (1) the capital costs of land acquisition and improvement, (2) the capital costs of facility construction, (3) other operating and maintenance costs, including license fees and taxes, (4) groundwater-quality restoration, surface reclamation, and plant decommissioning; and (5) research and development costs, including postmonitoring associated with potential future improvements of operation and maintenance. As in the case of benefits, the applicant should discount these costs to present worth.

There are also external costs. Their effects on the interests of people need to be examined. The applicant should supply, as applicable, an evaluation, including supporting data and rationale, regarding such external social and economic costs. For each cost, the applicant should describe the probable number and location of the population group adversely affected, the estimated economic and social impact, and any special measures to be taken to alleviate the impact.

Temporary external costs include shortages of housing; inflationary rentals or prices; congestion of local streets and highways; noise and temporary esthetic disturbances; overloading of water supply and sewage treatment facilities; crowding of local schools, hospitals, or other public facilities; overtaxing of community services; and disruption of people's lives or of the local community caused by acquisition of land for the proposed site.

Long-term external costs include impairment of recreational values (e.g., reduced availability of desired species of wildlife and sport animals, restrictions on access to land or water areas preferred for recreational use); deterioration of esthetic and scenic values; restrictions on access to areas of scenic, historic, or cultural interest; degradation of areas having historic, cultural, natural, or archeological value; removal of land from present or contemplated alternative uses; reduction of regional products because of displacement of persons from the land proposed for the site; lost income from recreation or tourism that may be impaired by environmental disturbances; lost income attributable to environmental degradation; decrease in real estate values in areas adjacent to the proposed facility; and increased costs to local governments for the services required by the permanently employed workers and their families. In discussing the costs, the applicant should indicate to the extent practical who is likely to be affected and for how long.

7.6.3 Resources Committed

Any irreversible and irretrievable commitments of resources due to the operation should be discussed. This discussion should include both direct commitments such as depletion of uranium resources and irreversible environmental losses such as destruction of wildlife habitat.

In this discussion, the applicant should consider lost resources from the viewpoints of both relative impacts and long-term net effects. As an example of relative impact assessment, the loss of a few animals of a given species could represent quite different degrees of significance, depending on the total population in the immediate region. Such a loss in the case of a small local population, however, could be less serious if the same species were abundant in neighboring regions. Similarly, the loss of a given area of highly desirable land should be evaluated in terms of the total amount of such land in the environs. These relative assessments should accordingly include statements expressed in percentage terms in which the amount of expected resource loss is related to the total resource in the immediate region and in which the total in the immediate region is related to that in surrounding regions. The latter should be specified in terms of areas and distances from the site.

8. ALTERNATIVES TO THE PROPOSED ACTION

In this chapter, the applicant's choice of the particular extraction and recovery processes for the ore body must be supported through a comparative evaluation of available alternatives. To the extent possible, the applicant should discuss realistic alternatives for the various processing stages. The NRC will consider all those alternatives that may reduce or avoid significant adverse environmental, social, and economic effects expected to result from construction and operation of the proposed activity. The NRC will not preselect the alternatives that should be considered by the applicant; rather, the applicant should make this determination for this specific case and should also make clear the bases and rationales for the choices in regard to number, availability, suitability, and factors limiting the range of alternatives that might avoid some or all of the environmental effects identified in Chapter 7, "Environmental Effects."

In the discussion of waste management alternatives, consideration should be given to the following siting, design, and operational performance objectives developed by the staff in addition to the postreclamation performance objectives indicated in Chapter 6:

1. Locate the liquid impoundment area(s) at sites remote from people in order to reduce potential exposures. They should also be located at sites where disruption and dispersion by natural forces are eliminated or reduced to the maximum extent reasonably achievable.

2. Design the impoundment area(s) so that seepage of toxic materials into the ground-water system would be eliminated or reduced to the maximum extent reasonably achievable.

9. BENEFIT-COST ANALYSIS

In this chapter, the applicant's benefit-cost statement should be summarized. The presentation should be made in the form of a narrative with accompanying tables and charts. It should clearly discuss the important benefits and costs of the proposed operations to justify the issuance of the license.

The applicant will have to develop criteria for assessing and comparing benefits and costs where these are expressed in nonmonetary or qualitative terms. The rationales for the selection of process alternatives as well as subsystem alternatives should be presented. In any case, the applicant should describe potential cumulative effects and should discuss in detail the tradeoffs that were made in order to warrant licensing of the proposed operation. The rationale for omitting apparent benefits or costs from the applicant's analysis should be explained. The applicant should key all the terms used in the benefit-cost analysis to the relevant sections of the application.

10. ENVIRONMENTAL APPROVALS AND CONSULTATIONS

List all licenses, permits, and other approvals of construction and operations required by Federal, State, local, and regional authorities for the protection of the environment.* List those Federal and State approvals that have already been received, and indicate the status of pending approvals. For general background, submit similar information regarding approvals, licenses, and contacts with local authorities.

Discuss the status of efforts to obtain a water quality certification under Section 401 and discharge permits under Section 402 of the Federal Water Pollution Control Act, as amended, if required. If not already obtained, indicate when certification is expected. If certification is not required, explain.

In view of the potential effects of the proposed operation on the economic development of the region in which it would be located, the applicant should also note the State, local, and regional planning authorities contacted or consulted. Office of Management and Budget Circular A-95** identifies the State, metropolitan, and regional clearinghouse that should be contacted, as appropriate.

Cite meetings held with environmental and other citizen groups with references to specific instances of the applicant's compliance with citizen group recommendations.

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[*] This list should be updated bimonthly when applications have been submitted but approvals have not as yet been received until final action is taken by the NRC.

[**] Inquiries concerning this circular may be addressed to the Office of Management and Budget, Washington, D.C. 20503.

11. REFERENCES

The applicant should provide a bibliography of all sources used in preparing the application. References cited should be keyed to the specific sections and page numbers to which they apply. The applicant should also list the names, together with their qualifications (expertise, experience, professional disciplines), of the persons who were primarily responsible for preparing the application.

DRAFT VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

An NRC source material license is required in order to process uranium solutions extracted from in situ uranium solution extraction operations. General guidance for filing an application is provided in §40.31 of 10 CFR Part 40, "Domestic Licensing of Source Material." Draft Regulatory Guide FP 818-4 provides specific guidance on the format and content of applications, including environmental reports, for licenses to authorize commercial-scale in situ uranium solution extraction operations. The guide conforms to NRC regulations and reflects experience gained over the past 5 years in actual licensing cases.

1.2 Need for Proposed Action

Basic detailed guidance is essential to applicants for the efficient preparation of applications, including environmental reports, for commercial-scale in situ uranium solution extraction facility licenses and for their review by the NRC staff. Such guidance is presently delineated neither in NRC regulations nor in regulatory guides. Based on experience gained in the issuance of such licenses, the NRC staff has identified information that should be contained in applications (including environmental reports) to reflect present needs and practices. Such information in a regulatory guide will be helpful to both applicants and the NRC staff in reducing the cost and time involved in preparing and processing license applications.

1.3 Value/Impact of Proposed Action

1.3.1 NRC

The proposed guide describes the information needed from applicants for commercial-scale in situ uranium solution extraction facility license reviews by the NRC. It identifies present needs and reflects present practices. It should result in a significant reduction in the number of questions and requests for clarification submitted by the NRC staff to applicants, improve the consistency in application reviews because of more uniform application submittals, reduce the amount of staff review effort in license processing, and expedite licensing actions.

1.3.2 Other Government Agencies

Not applicable, unless the government agency is an applicant such as TVA.

1.3.3 Industry

The value/impact on applicants and licensees will be essentially the same as indicated above.

1.3.4 Public

Since there is no reduction or relaxation of any safety requirements, no impact on the public is anticipated other than an improved public understanding of information considered by the NRC staff in the licensing process.

1.4 Decision on Proposed Action

As indicated in Section 1.2, specific guidance on the format and content of applications (including environmental reports) for licenses to authorize commercial-scale in situ uranium solution extraction should be clearly delineated. Thus, Draft Regulatory Guide FP 818-4 should be issued.

2. TECHNICAL APPROACH

2.1 Technical Alternatives

None available.

2.2 Discussion and Comparison of Technical Alternatives

Not applicable.

2.3 Decision on Technical Approach

Not applicable.

3. PROCEDURAL APPROACH

3.1 Procedures Alternatives

Possible SD procedures that may be used to carry out the proposed action include the following:

- Regulation
- Regulatory guide
- ANSI standard, endorsed by a regulatory guide
- NUREG report
- Branch position

3.2 Value/Impact of Procedural Alternatives

A regulation does not appear to be a reasonable alternative because guidance, rather than specific regulatory requirements, is needed. A NUREG report does not appear appropriate since a NUREG report usually contains only results of specific studies and/or basic research findings. No specific branch position has been prepared, nor is one anticipated. Preparation of a regulatory guide appears to be the most viable course of action.

3.3 Decision on Procedural Approach

A regulatory guide should be prepared delineating specific guidance on the format and content of applications (including environmental reports) for licenses to authorize commercial-scale in situ uranium solution extraction operations.

4. STATUTORY CONSIDERATIONS

4.1 NRC Authority

NRC authority is delineated in the proposed guide.

4.2 Need for NEPA Assessment

The proposed action is not a major action, as defined in 10 CFR Part 51, and does not require an environmental impact statement.

5. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICIES

Not applicable. The guide is intended to provide instructive guidance for commercial-scale in situ uranium solution extraction license applicants. It is not considered as a substitute for a careful evaluation of the proposed program by an applicant. No backfitting is expected from its issuance.

6. SUMMARY AND CONCLUSIONS

A regulatory guide should be prepared.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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OFFICE OF STANDARDS DEVELOPMENT

DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

August 1980
Division 8
Task OH 941-4

Contact: H. Pettengill (301) 443-5970

INFORMATION RELEVANT TO ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES AT URANIUM MILLS WILL BE AS LOW AS IS REASONABLY ACHIEVABLE

A. INTRODUCTION

Paragraph 20.1(c) of 10 CFR Part 20, "Standards for Protection Against Radiation," states that licensees should make every reasonable effort to keep radiation exposures, as well as releases of radioactive material to unrestricted areas, as far below the limits specified in Part 20 as is reasonably achievable. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," sets forth the philosophy and general management policies and programs that licensees should follow to achieve this objective of maintaining radiation exposures to employees "as low as is reasonably achievable" (ALARA).

This guide is directed specifically toward uranium mill licensees and recommends design criteria and administrative practices acceptable to the NRC staff for maintaining occupational exposures ALARA in uranium mills. However, since the basic processes at other types of uranium recovery facilities have a similar potential for exposing workers to uranium and its daughters, the guidance provided in this guide can be applied to those facilities as well. An existing NRC report, NUREG-0511, "Draft Generic Environmental Impact Statement on Uranium Milling" (Ref. 1), also provides detailed information for controlling the radiation hazard and chemical toxicity of airborne uranium and its daughter products in uranium mills.

This guide is generally directed toward occupational health protection from radiologic and toxic hazards from airborne particulates of uranium and its daughters; however, it is also recognized that uranium mill workers will

FOR COMMENT

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by OCT 1 [illegible]

Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

be exposed to external radiation in addition to inhaled particulates. Therefore, the need to ensure protection of mill workers from external radiation hazards is also advised.

Specific guidance regarding protection from radiologic and toxic hazards caused by materials in effluents to unrestricted areas is beyond the scope of this guide. This topic is mentioned only in connection with actions that influence both occupational exposure and effluent control. However, the same controls that have been shown to keep occupational exposures to airborne uranium and its daughters ALARA also tend to keep releases of these materials from the mill ALARA.

B. DISCUSSION

The principle of maintaining occupational radiation exposures as low as is reasonably achievable is an extension of an original recommendation of the National Committee on Radiation Protection (NCRP) (now the National Council on Radiation Protection and Measurements) in its 1949 report (published in 1954 as Report No. 17 (Ref. 2)). In this early report, the NCRP introduced the philosophy of assuming that any radiation exposure may carry some risk and recommended that radiation exposure be kept at a level "as low as practicable" (currently referred to as ALARA) below the recommended maximum permissible dose equivalent. Similar recommendations to keep exposures ALARA have been included in NCRP reports up to the present time (Ref. 3), as well as in recommendations of the National Academy of Sciences--National Research Council (Ref. 4), the Federal Radiation Council (Ref. 5), and other independent scientific and professional organizations (Ref. 6). The basic radiation protection philosophy of these recommendations has been incorporated in regulations and guides of the Nuclear Regulatory Commission.

This guide provides a very detailed supplement for uranium mill licensees of the basic philosophy of Regulatory Guide 8.10, which lists for all specific licensees the types of management commitments and radiation protection programs that would help to achieve the objective of maintaining occupational exposures ALARA.

Regulatory Guide 3.5, "Standard Format and Content of License Applications for Uranium Mills," outlines the information that applicants should include in their application for a uranium mill license; however, it does not outline the detailed information that would be required to establish acceptable health physics and ALARA programs. This regulatory guide describes the details of an acceptable health physics and ALARA program that an applicant would describe as recommended in Section C.5, "Operations," of Regulatory Guide 3.5.

This guide; Regulatory Guide 3.5; Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection"; Regulatory Guide 8.22, "Bioassay at Uranium Mills"; and Draft Regulatory Guide OH 710-4, "Health Physics Surveys in Uranium Mills," will be used as the bases for evaluating license applications and radiation safety programs of NRC-licensed uranium mills.

C. REGULATORY POSITION

The principles and practices presented in this guide should be used as guidance in developing the health physics and ALARA programs for a uranium mill for appropriate sections of an application* for a new or renewal license. The recommendations of this guide are intended to assist the applicants in preparing license applications that are acceptable to the NRC licensing staff and are consistent with the philosophy of ALARA. Unique features not addressed here will require specific review by the NRC licensing staff.

A licensee's program for occupational protection against uranium and its daughters will be considered consistent with the ALARA philosophy if the uranium mill's operating policies and programs satisfy the following major principles and practices.

1. ALARA PHILOSOPHY

The purpose of the radiation protection program at a uranium mill is to maintain radiation exposure ALARA for all employees, contractors, visitors, and members of the general public. Thus, the implementation of a successful ALARA program becomes the responsibility of everyone incidental to the processing of uranium ores. Responsibilities for conducting an ALARA program are shared by licensee management,* the radiation safety officer (RSO),** and all mill workers.

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* An application and a suggested format for its completion may be obtained from the licensing staff of the Division of Waste Management, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

1.1 Licensee Management

Licensee management should provide the following:

1. Information and policy statements to employees, contractors, and visitors;
2. Periodic management audits of procedural and operational efforts to maintain exposures ALARA;
3. Continuing management evaluation of the health physics program, its staff, and its allocation of adequate space and money;
4. Appropriate briefings and training in radiation safety, including ALARA concepts for all uranium mill employees and, when appropriate, for contractors and visitors.

1.2 Radiation Safety Officer

The Radiation Safety Officer (RSO) should be delegated the following:

1. Sufficient authority to enforce regulations and administrative policies that affect any aspect of the radiological safety program;
2. Responsibility to develop and administer the ALARA program;
3. Authority to review and approve plans for new equipment, process changes, or changes in operating procedures to ensure that the plans do not adversely affect the protection program against uranium and its daughters.

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*"Management" is defined here as those persons authorized by the licensee of record to make policies and to direct activities of the recovery facility.

**The title "Radiation Safety Officer" is used synonymously with radiation protection manager by many licensees and will be used in this guide to designate the qualified individual who is responsible for developing and supervising the radiation safety program; other titles are equally acceptable.

1.3 Mill Workers

All workers at the mill should be responsible for the following:

1. Adhering to all rules, notices, and operating procedures for radiation safety established by licensee management and the RSO;
2. Reporting promptly to the RSO and licensee management the witness of equipment malfunction or the violation of standard practices or procedures that could result in increased radiological hazard to any individual;
3. Suggesting improvements for the ALARA program.

2. HEALTH PHYSICS ORGANIZATION AND ADMINISTRATIVE PROCEDURES

2.1 Health Physics Authorities and Responsibilities

The radiation safety officer at the mill site should be responsible for conducting the health physics program and for assisting the resident manager in ensuring compliance with NRC's regulations and the license conditions applicable to worker health protection.

Generally, the RSO should report directly to the resident manager on matters of safety. The RSO should be directly responsible for supervising the health physics technicians, for overseeing the day-to-day operation of the health physics program, and for ensuring that records required by the NRC are maintained. The RSO should have both the responsibility and the authority to suspend, postpone, or modify any work activity that is potentially hazardous to workers or a violation of the Commission's regulations or license conditions. The RSO may have other safety-related duties, such as responsibility for programs of industrial hygiene and fire safety, but should have no direct production-related responsibility.

2.2 Operating Procedures

Standard written operating procedures should be established for all operational activities involving radioactive materials that are handled, processed, or stored. Standard operating procedures for operational activities should include a consideration of pertinent radiation safety practices. Additionally, written procedures should also be established for nonoperational activities, to include health physics and environmental monitoring, sampling, analysis, and instrument calibration. An up-to-date copy of each written procedure should be kept in each area where it is used.

All written procedures for both operational and nonoperational activities should be reviewed and approved in writing by the RSO before being implemented and whenever a change in a procedure is proposed to ensure that proper radiation protection principles are being applied. In addition, the RSO should review all existing operating procedures at least annually to ensure the procedures do not violate any newly established radiation protection practices.

For work or nonroutine maintenance jobs where the potential for exposure to radioactive material exists and for which no standard written operating procedure already exists, a radiation work permit (RWP)* should be used. Such permits should describe the following:

1. The details of the job to be performed.
2. Any precautions necessary to reduce exposure to uranium and its daughters.
3. The radiological monitoring and sampling necessary during and following completion of the job.

The RSO should indicate by signature the review of each RWP prior to the initiation of work, and the work should be carried out in strict adherence to the conditions of the RWP. When the RSO is not available, e.g., during off-shifts, the RSO should designate a member of the radiation safety office staff or a supervisory member of the production staff who has received specialized radiation protection training to review and sign RWPs in the RSO's absence.

2.3 Surveillance: Audits and Inspections

It has been observed repeatedly that where sufficient management interest exists, exposure to hazardous materials is reduced. Frequent management audit and inspection of worker health protection practices at a uranium mill can serve to provide management with the information necessary to conduct an appropriate ALARA program.

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*The term, "radiation work permit" is used by many licensees and will be used throughout this guide; other terms, such as special work permit, are equally acceptable.

2.3.1 Daily and Weekly Inspections

The RSO or designated health physics technician and the mill foreman should conduct a daily walk-through (visual) inspection of all areas of the mill to ensure proper implementation of good safety practices, including good housekeeping and cleanup practices that would minimize unnecessary contamination and ensure adherence to standard operating procedures. Problems observed should be noted in writing in a daily inspections' logbook. The entries should be dated, signed, and maintained on file for at least 1 year. The RSO should review each day's findings of violations of radiation safety procedures or other potentially hazardous problems with the resident manager and other mill employees who have authority to correct the problem. Also, the RSO should review the daily work-order and shift logs on a regular basis to determine that all jobs and operations having a potential for exposing personnel to uranium, especially those jobs that would require a radiation survey and monitoring, were approved in writing by the RSO or his staff prior to initiation of work.

A weekly inspection should be made by the health physics technician of all work and storage areas and a report submitted to the RSO on any items of non-compliance with operating procedures, license requirements, or safety practices affecting radiological safety.

2.3.2 Monthly Inspections

At least monthly, the RSO should conduct an inspection of all work and storage areas and should review all monitoring and exposure data for the month. The RSO should provide to the resident manager and all department heads for their review a written summary of the month's significant worker protection activities containing (1) a summary of personnel exposure data, including bioassays and time-weighted calculations, and (2) a summary of all pertinent radiation survey records.

In addition, the monthly inspection summary should specifically address any trends or deviations from the ALARA program, including an evaluation of the adequacy of the implementation of license conditions regarding ALARA. The summary should provide a description of unresolved problems and the proposed corrective measures. Monthly summary inspection reports should be maintained on file and readily accessible for at least 5 years.

2.3.3 ALARA Program Audit

The RSO should perform a formal semiannual audit of the ALARA program and submit a detailed written report on the audit to the resident manager. The primary purpose of the audit is to evaluate the overall effectiveness of the mill ALARA program. The audit report should summarize the results of the following data:

1. Employee exposure records (external and time-weighted calculations)
2. Bioassay results
3. Inspection log entries and summary reports of daily, weekly, and monthly inspections
4. Documented training program activities
5. Safety meeting reports
6. Radiological survey and sampling data
7. Radioactive effluent and environmental monitoring data
8. Reports on overexposure of workers submitted to NRC, MSHA, or States
9. Operating procedures that were reviewed during this time period.

The report on the semiannual ALARA audit should specifically discuss the following:

1. Trends in personnel exposures for identifiable categories of workers and types of operational activities
2. Trends in effluent releases
3. Whether equipment for exposure control and effluent control is being properly used, maintained, and inspected
4. Recommendations on ways to further reduce personnel exposures and effluent releases of uranium and its daughters.

2.4 Technical Qualifications of Health Physics Staff

2.4.1 Radiation Safety Officer

The RSO should have the following education, training, and experience:

1. Education: A bachelor's degree in the physical sciences or engineering from an accredited college or university.
2. General experience: One year of supervisory experience and one year of experience in a uranium mill or related industry.

3. Health physics experience: One year of work experience in applied health physics, radiation protection, industrial hygiene, or similar work. This experience should involve actually working with radiation detection and measurement equipment rather than only administrative or "desk" work.

4. Specialized training: A formalized intensive course in health physics of at least 4 weeks' duration. At least 1 week of the course should be specifically applicable to health physics for uranium milling and mining, if both are applicable. In addition, the RSO should attend a refresher course on uranium mill health physics every 2 years.

5. Specialized knowledge: A thorough knowledge of the proper application and use of all health physics equipment used in the mill, the chemical and analytical procedures used for radiological sampling and monitoring, and methodologies used to calculate personnel exposure to uranium and its daughters.

2.4.2 Health Physics Technicians

In addition to the RSO, there should be a minimum of one full time health physics technician at every uranium mill. The health physics technician should have the following education, training, and experience:

1. Education: An associate degree in the phycical sciences, engineering, or a health-related field. Alternatively, a high school diploma plus 2 years of relevant work experience in applied radiation protection are acceptable.

2. General experience: One year of previous work experience in a uranium mill or related industry involving radiation protection.

3. Health physics experience: One year of work experience using sampling and analytical laboratory procedures that involve health physics, industrial hygiene, or industrial safety measures to be applied in a uranium mill.

4. Specialized training: At least 4 weeks of formalized training in radiation health protection for uranium mills.

5. Specialized knowledge: A working knowledge of the proper operation of health physics instruments used in the mill, surveying and sampling techniques, and personnel dosimetry requirements.

2.5 Radiation Safety Training

All new employees should be instructed by means of an established course in the inherent risks of exposure to radiation and the fundamentals of protection against exposure to uranium and its daughters before beginning their jobs. Other guidance pertinent to this course is found in Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure," and Draft Regulatory Guide OH 902-1, "Instruction Concerning Risk from Occupational Radiation Exposure." This course of instruction should include the following topics:

1. Fundamentals of Health Protection
    - a. What are the radiologic and toxic hazards of exposure to uranium and its daughters
    - b. How uranium and its daughters enter the body (inhalation and ingestion)
    - c. Why exposures to uranium and its daughters should be kept as low as is reasonably achievable (ALARA)
2. Personal Hygiene at Uranium Mills
    - a. Wearing protective clothing
    - b. Using respirators, when appropriate
    - c. Eating, drinking, and smoking only in designated areas
    - d. Using proper methods for decontamination (i.e., showers).
3. Facility-Provided Protection
    - a. Cleanliness of the work place
    - b. Safety-designed features for process equipment
    - c. Ventilation systems and effluent controls
    - d. Standard operating procedures
    - e. Security and access control to designated areas.
4. Health Protection Measurements
    - a. Measurement of airborne radioactive materials
    - b. Bioassays to detect uranium (urinalysis and *in vivo* counting)
    - c. Surveys to detect contamination of personnel and equipment
    - d. Personnel dosimetry
5. Radiation Protection Regulations
    - a. Regulatory authority of NRC, MSHA, and State
    - b. Employee rights in 10 CFR Part 19
6. Mill Emergency Procedures.

A written test with questions directly relevant to the principles of radiation safety and health protection in uranium milling covered in the training course should be given to each worker. The instructor should review the test results with each worker. The instructor should discuss any wrong answers to test questions with the worker until the worker understands the correct answer. Workers who fail the test should be retested. These tests and results should be maintained on file. Each permanent worker should be provided an abbreviated retraining course annually. Documented successful completion of the retraining course should also be maintained on file. Retraining should include relevant information that has become available during the past year, a review of safety problems that have arisen during the year, changes in regulations and license conditions, exposure trends, and other current topics.

In addition, all new workers, including supervisors, should be given specialized instruction on the health and safety aspects of the specific jobs they will perform. This instruction should be in the form of individualized on-the-job training. Supervisors should be provided additional specialized training on their supervisory responsibilities in the area of worker radiation protection. Retraining should be conducted annually and documented. All employees should sign a statement that they received job-specific safety training. The statement should indicate the dates the training was received and it should be cosigned by the instructor. Also, every 2 months all workers should attend a general mill safety meeting with at least 30 minutes of the meeting devoted to radiation safety matters.

All visitors who have not received training should be escorted by someone properly trained and knowledgeable about the hazards of the mill. As a minimum, visitors should be instructed specifically on what they should do to avoid possible hazards in the areas of the mill they will be visiting.

Contractors having work assignments in the mill should also be given appropriate training and safety instruction. Contract workers who will perform work on heavily contaminated equipment should receive the same training and safety instruction normally required of all permanent workers. Only job-specific safety instruction is necessary for contract workers who have previously received full training on prior work assignments at the mill.

2.6 Surveys

The RSO and radiation safety office staff are responsible for performing all routine and special radiation surveys as required by license conditions and 10 CFR Part 20. Acceptable survey methods are specified in Section C.1 of Draft Regulatory Guide OH 710-4, "Health Physics Surveys in Uranium Mills."

2.7 Respiratory Protection

The RSO and the radiation safety office staff are responsible for the implementation of a respiratory protection program. There should be adequate supplies of respiratory devices to enable assignment of a device to each individual who may routinely enter airborne radioactivity areas. Additional respiratory protection devices should be located near access points of airborne radioactivity areas. All airborne radioactivity areas should have controlled access. Routine physical (medical) evaluation should be required of those individuals who will use respirators. As a minimum, the respiratory protection program should meet the recommendations in Regulatory Guide 8.15 and as supported in NUREG-0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials" (Ref. 7).

2.8 Bioassay Procedures

The RSO is responsible for implementing a bioassay program. The frequency adopted and the type of analysis should meet the recommendations in Regulatory Guide 8.22.

3. FACILITY AND EQUIPMENT DESIGN

General considerations for the design of uranium mills and uranium ore processing equipment should not be based solely on chemical process efficiency, but should also be based on the relative potential for radiologic and toxic hazards resulting from exposure of personnel to uranium and its daughters. Major aspects of planning and design that should be considered are discussed below.

3.1 Space Layout

Facility layout should be designed to maintain employee exposures ALARA while at the same time ensuring that exposure to other persons is not thereby increased. General provisions of the mill layout should include the following considerations:

1. The need for access to process equipment and the need to perform routine maintenance.
2. The need to maintain adequate ventilation in all mill areas in which radioactive materials might be spilled, suspended, or volatilized;
3. Provisions to enable isolation of yellowcake drying, packaging, and shipping areas from other mill process areas;
4. Provisions for controlling access to the uranium mill proper and the ability to secure or restrict entry to any airborne radioactivity area; and
5. The need to locate emergency personnel decontamination equipment (e.g., shower facilities) adjacent to mill equipment that, in the advent of an accident, a spill, or equipment malfunction, could cause gross contamination of a worker.

3.2 Access Control

Access to airborne radioactivity areas should be controlled or restricted by the use of caution signs, operational procedures, or security locks.

3.3 Ventilation Systems

To the extent practicable, accomplish the following:

1. Provide local exhaust ventilation (such as chemical hoods) or general area ventilation where concentrations of natural uranium may be present in excess of 10% of the values given in Table 1 of Appendix B to 10 CFR Part 20. The design ventilation rate (air exchange rate) should be sufficient to maintain airborne concentrations of natural uranium to less than 10% of the maximum permissible concentration (MPC) given in Table 1 of Appendix B to 10 CFR Part 20.
2. Design exhaust stacks so that exhausted air will not enter air intakes that service other mill areas.

3. Locate exhaust vents in a way that ensures compliance with the requirements of § 20.106 of 10 CFR Part 20 and 40 CFR Part 190 for effluents to unrestricted areas, as well as ALARA exposure considerations for the worker and the general public.

4. Where approriate, include specific types of filters or air scrubbers for the exhaust air to minimize the release of uranium to the environs.

3.4 Fire Control and Chemical Hazard Protection Systems

The mill design should provide for the isolation of mill areas where there is a high potential for fire and where uranium could be dispersed as the result of a fire. Provisions should be made for fire alarms, fire extinguishers, sprinkler systems, fire hydrants, water tanks, and other general fire fighting equipment. Emergency procedures and training should include immediate fire control as a priority item. Design features should include automatic fire suppression and detection equipment in high fire-potential areas (i.e., solvent extraction area). In the event of fire, there should be provision for drainage of solvent to sumps or to outside lined ponds. Appropriate caution signs should be posted in areas of fire hazard. Fire detection systems should be checked weekly. Fire drills should be performed at least semiannually.

Facility design should provide for the construction of dikes and curbs around process and storage tanks to confine hazardous chemicals in the event of a spill or leak. Tanks containing hazardous chemicals should be equipped with high-level alarms to minimize the possibility of spillage caused by a malfunction of the process circuit.

3.5 Laboratory Design Features

Consideration should be given to providing different laboratory facilities for metalurgical and bioassay analyses, if they are both performed at mill site. Owing to the sensitivity required in performing bioassay analyses, provisions should be made to ensure against cross-contamination of uranium from mill ore samples. Laboratory equipment and surfaces should be constructed of materials that are easily decontaminated. Laboratory surfaces used for the preparation of bioassay samples should be decontaminated daily to less than 200 dpm $\alpha/100cm^2$ of total surface contamination. All mill laboratories should provide adequate general ventilation and exhaust fume hoods when appropriate. Special attention should be directed to the design of air exhaust systems that service ore sample pulverizing and grinding equipment. The design of the laboratory should provide for the safe handling, storage, and disposal of radioactive wastes resulting from sample analyses.

3.6 Ore and Product Storage

Uranium mill plans should include the following areas:

1. Provide for raw ore storage, fine ore bins, and yellowcake storage in areas such that they do not cause unnecessary exposure to mill personnel.
2. Provide adequate space in the yellowcake packaging area to conduct an initial survey and smear test of each yellowcake package and to enable decontamination of drums to avoid transporting a contaminated package through other mill areas.
3. Locate yellowcake storage and shipping areas so as to minimize the handling time required prior to shipment.

3.7 General Equipment Considerations

General features applicable to equipment that will be used for handling, containing, or contacting uranium and its daughters are as follows:

1. Equipment that contains large volumes of uranium-bearing liquids should be designed with sumps or dikes to contain the liquids in the advent of leaks or spills.
2. Equipment should be designed to optimize the ease of carrying out procedures, especially routine maintenance, thereby minimizing working time where personnel are exposed to radiation and maximizing distances of personnel from the source of radiation with which they are working, consistent with the purposes of the procedures.
3. Appropriate caution signs and symbols should be provided to meet the requirement of § 20.203 of 10 CFR Part 20.

4. CONTROL OF AIRBORNE URANIUM AND ITS DAUGHTERS

One of the major inhalation hazards associated with uranium milling facilities results from the resuspension in air of uranium and its daughters. Therefore, properly designed ventilation and dust control systems are needed to ensure that exposure of workers is maintained ALARA. There are, in general, four areas that present radiologic and toxic hazards caused by airborne materials at a typical mill. These areas encompass (1) ore storage, handling, and crushing, (2) ore grinding, leaching, and concentrating processes, (3) ore precipitation, drying, and packaging, and (4) miscellaneous mill locations. Appropriate design objectives for ventilation and dust control systems recommended for each of these generalized mill areas are given below.

4.1 Ore Storage, Handling, and Crushing Areas

Where ore is handled in the open, the objective should be to minimize blowing of dust. Water sprinkling systems are recommended for use on ore piles when the ore moisture content is less than 10%. If ore is crushed and transported in the dry state, (i.e., moisture content less than 25%) the use of ventilation systems and dust collectors are recommended. As ore travels along conveyor belts to the grinder, all drop points should have either hooded dust collectors or dust supressant systems, such as sprinklers or foam ejectors. When crushers are used prior to grinding, it is recommended that a hooded ventilation system be installed over all external openings to the crusher. The use of wet scrubbers or dust collectors is recommended for ventilation systems that service ore storage, handling, and crushing areas of the mill.

4.2 Grinding, Leaching, and Concentrating Process Areas

Although many methods of ore grinding exist, a preferred method is the semiautogenous method. Effectively, the semiautogenous grinding method is a one-step process that eliminates the need for crushers. Thus, the need for most dust collectors normally required at mills using crushers and rod or ball grinders would be negated. However, general ventilation systems are recommended to service mill areas where any grinding method is performed to ensure against the buildup of radon and its daughters normally released in the grinding process.

The ventilation rate should be adequate to maintain the concentration of natural uranium to less than 10% of the concentration values specified in paragraph 4.a of the note to Appendix B to 10 CFR Part 20, (i.e., 7.5 μgm/m^3 of air or 5 x 10^{-12} μCi/cm^3). It is recommended that all leaching and thickening tanks located in enclosed structures be covered and vented directly to the outside atmosphere. General ventilation systems for mill areas where leaching and thickening tanks are located should be designed to maintain natural uranium ore dust concentrations in air at less than 7.5 μgm/m^3. If the mill is so designed that the solvent extraction (SX) concentration process equipment is in enclosed structures, a general ventilation system is recommended and should be designed to maintain the airborne uranium concentration in air to less than 20 μgm/m^3 or 1 x 10^{-11} μCi/cm^3, (i.e., 10% of the MPC for uranium other than natural ore dust). The use of wet scrubbers on general ventilation systems that service areas of the mill where grinding and leaching equipment are located is recommended. Scrubbers are not necessary on ventilation systems that service areas of the mill where the clarification or solvent extraction equipment is located.

4.3 Precipitation, Drying, and Packaging Areas

During the precipitation step of the uranium recovery process, the uranium-bearing solution is neutralized with ammonia to produce ammonium diuranate, (yellowcake) which is then thickened by dewatering. Some yellowcake and ammonia gas may be released into the air because of the agitation of this solution. General ventilation systems are required and should be designed to maintain the concentration in air of yellowcake near precipitation tanks, yellowcake thickeners, yellowcake filters, and yellowcake repulp equipment to less than 20 μgm/m^3 or 1 x 10^{-11} μCi/cm^3 (10% of the MPC). The next step of the recovery process involves the drying and packaging of yellowcake. Since the potential for the release of airborne yellowcake is much greater in dry form, it is recommended that drying and packaging of yellowcake should be performed in an enclosure that is separated from other areas of the mill. Also, the drying and packaging enclosure should be maintained under negative pressure. A separate air suction ring system should also be used at each yellowcake drumming station. Individual suction ring systems need only be operated during periods when the drum at that location is being filled. The exhausts for the drying and packaging enclosure and the suction ring should be vented through a wet scrubber. To ensure proper operation, scrubber circuits from the concentrate drying and packaging areas should be checked every hour and documented. Manometer readings should be recorded once per shift and subsequently documented.

4.4 Miscellaneous Locations

Other important areas of the mill that have the potential for containing hazardous levels of uranium and its daughters in air include maintenance shops, rubber shops, metallurgical and bioassay laboratories, and general laundries, if they exist. Each of the above mill areas should be serviced by ventilation systems designed to maintain air concentration of uranium to less than 20 $\mu gm/m^3$ or 1×10^{-11} $\mu Ci/cm^3$. Wet scrubbers are not necessary on these systems, however, bag filters are recommended.

REFERENCES

1. "Draft Generic Environmental Impact Statement on Uranium Milling," NRC report NUREG-0511, April 1979.

2. National Bureau of Standards, "Permissible Dose from External Sources of Ionizing Radiation," Handbook 59, Recommendations of the National Council on Radiation Protection (NCRP Report No. 17), Washington, D.C., September 24, 1954.

3. National Council on Radiation Protection and Measurements, "Review of the Current State of Radiation Protection Philosophy," Report No. 43, Washington, D.C., January 15, 1975.

4. National Academy of Sciences - National Research Council, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," Washington, D.C., 1972.

5. Federal Radiation Council, "Background Material for the Development of Radiation Protection Standards," Report No. 1, Washington, D.C., 1960.

6. International Commission on Radiological Protection, "Implications of Commission Recommendations That Doses Be Kept As Low As Readily Achievable," Report No. 22, Pergamon Press, Elmsford, New York, 1974.

7. "Manual of Respiratory Protection Against Airborne Radioactive Materials," NRC report NUREG-0041, October 1976.

DRAFT VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

Applicants for a uranium milling license must submit a license application containing the information specified in Regulatory Guide 3.5, "Standard Format and Content of License Applications for Uranium Mills." The purpose of this proposed action is to describe both administrative health physics programs and methods to achieve ALARA occupational exposure to workers that are acceptable to the NRC staff. Health physics programs are covered in Section C.5, "Operations," in Regulatory Guide 3.5.

1.2 Need

Now, licensees are uncertain what the NRC staff will accept in the way of a health physics program or procedures and design features needed to achieve ALARA exposures in a uranium mill. As a consequence, a wide variety of programs are submitted. In order to meet minimum standards, much correspondence between the applicant and NRC is required. A guide will reduce the amount of correspondence needed, save manpower for both NRC and the applicant, show clearly how NRC regulations apply to uranium mills, and establish a uniform standard for an acceptable health physics and ALARA program for worker protection.

1.3 Value/Impact

1.3.1 NRC

The impact of the proposed guidance will be primarily to reduce licensing staff effort in reviewing applications and in corresponding with applicants about areas where the application does not meet NRC licensing practice. An estimated 0.75 man-year is required to develop the guide.

1.3.2 Other Government Agencies

The proposed guidance will impact on the Mine Safety and Health Administration (MSHA) because they also regulate occupational health protection at uranium mills and on Agreement State regulatory agencies that regulate mills, primarily New Mexico, Colorado, Texas, Washington, and Florida. A Memorandum of Understanding (MOU) signed by NRC and MSHA states that each agency will coordinate the development of standards with the other agency. The MOU was published in the Federal Register (45 FR 1315) on January 4, 1980.

1.3.3 Industry

Industry will benefit from having clear guidance on what constitutes NRC licensing policy. Some minor expense may be involved, however, in upgrading current health physics programs and in establishing an effective ALARA program where one does not currently exist to meet the recommendations in the guidance.

1.3.4 Workers

Workers' protection should improve from having clearly stated and consistent standards for health physics and ALARA programs. Workers and their representatives will now have access to a clearly defined standard ALARA program for uranium mills. This will help them understand whether their employer has an adequate program and why some things are done as they are.

1.3.5 Public

The guidance pertains to worker protection programs. It will not directly affect the public.

1.4 Decision

The NRC should develop guidance on a standard administrative health physics and ALARA program for worker protection that is acceptable to the NRC licensing staff.

2. TECHNICAL APPROACH

The technical approach in the guidance will be based on (1) NRC licensing policy as expressed in Safety Evaluation Reports (SER) written by the NRC licensing staff, especially the recent SER for Minerals Exploration Company Sweetwater Uranium Project and (2) other references to be cited in the guidance.

3. PROCEDURAL APPROACH

3.1 Procedural Alternatives

There are three reasonable procedural alternatives: (1) put the guidance directly in the regulations, (2) put the guidance in a regulatory guide, and (3) continue to handle on a case-by-case basis for each licensing application.

3.2 Value/Impact of Procedural Alternatives

Putting the guidance in the regulations is not really suitable for the type of guidance envisioned because some of the program must be tailored to the design and needs of the individual mill.

The guidance envisioned seems best suited for a regulatory guide. A guide provides the best mix of flexibility and clear statement of a uniform and consistent licensing policy.

3.3 Decision on Procedural Approach

Developing a regulatory guide is the favored procedural approach.

4. CONCLUSION

In summary, it is proposed that a regulatory guide on health physics and ALARA programs in uranium mills for worker protection be developed.

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DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

August 1980
Division 8
Task OH 710-4

Contact: S. A. McGuire (301) 443-5970

HEALTH PHYSICS SURVEYS IN URANIUM MILLS

A. INTRODUCTION

Section 40.32, "General Requirements for Issuance of Specific Licenses," of 10 CFR Part 40 of the Commission's regulations states that the Commission will approve an application for a specific license to operate a uranium mill if the applicant is qualified by reason of training and experience to use the source material for the purpose requested in such manner as to protect health and minimize danger to life and property and if the applicant's proposed equipment, facilities, and procedures are adequate to protect health and minimize danger to life and property.

The following sections of 10 CFR Part 20, "Standards for Protection Against Radiation," of the Commission's regulations deal with the protection of mill workers: § 20.201, which requires adequate surveys, § 20.101, which limits worker exposure to external radiation, § 20.103, which limits exposure to airborne radioactive material in restricted areas, § 20.202, which requires personnel monitoring for radiation in certain instances, § 20.203, which requires posting of warning signs and controlled access to areas with high radiation levels, § 20.401, which requires records of radiation surveys and personnel monitoring reports, and § 20.405, which requires reports of overexposures.

This guide describes health physics surveys acceptable to the NRC staff for protecting uranium mill workers from radiation and the chemical toxicity of uranium while on the job. The guidance can also be applied, in part, to other types of uranium recovery facilities since many of the processes used in uranium recovery facilities that have the potential for exposing employees to radioactivity are similar to operations conducted at uranium mills. The guide

FOR COMMENT

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by OCT 20 1980

Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

does not cover surveys to prevent the release of radioactive material to unrestricted areas or surveys to measure exposure of the public to radioactive materials in effluents, with the exception of surveys of the skin and clothing of workers leaving the mill and surveys of equipment and packages leaving the mill.

B. DISCUSSION

Regulatory Guide 3.5, "Standard Format and Content of License Applications for Uranium Mills," outlines the type of information that applicants for a uranium mill license should include in their application and suggests a uniform format for presenting that information. This regulatory guide describes health physics (radiation protection) surveys acceptable to the NRC licensing staff that an applicant may describe as recommended in Section C.5, "Operations," in Regulatory Guide 3.5.

The contents of this guide are based to a significant extent on the health physics survey programs now being required by the NRC licensing staff. The contents of this guide are also based to a large extent on the International Atomic Energy Agency (IAEA) "Manual of Radiological Safety in Uranium and Thorium Mines and Mills" (Ref. 1).

The subjects of respiratory protection and uranium bioassay are not included in this guide. Those subjects are covered in Regulatory Guides 8.15, "Acceptable Programs for Respiratory Protection," 8.22, "Bioassay at Uranium Mills," and Draft Regulatory Guide OH 941-4, "Information Relevant to Ensuring that Occupational Radiation Exposures at Uranium Mills Are As Low As Is Reasonably Achievable." Guidance on quality assurance programs can also be obtained from Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations)--Effluent Streams and the Environment."

C. REGULATORY POSITION

1. SURVEYS

1.1 Surveys for Airborne Uranium Ore Dust

Surveys for airborne uranium ore dust are necessary (1) to demonstrate compliance with the weekly intake limits for workers specified in paragraphs

20.103(b)(2) as well as the quarterly intake limits for workers specified in paragraph 20.103(a) of 10 CFR Part 20; (2) to meet the posting requirements for airborne radioactivity areas in § 20.203(d); (3) to determine whether precautionary procedures such as process or other engineering controls, increased surveillance, limitation on working times, provision of respiratory protective equipment, or other precautions should be considered to meet paragraphs 20.103(b)(1) and (b)(2); and (4) to determine whether exposures to radioactive materials are being maintained as low as is reasonably achievable as stated in paragraphs 20.1(c) and 20.103(b)(2).

The concentration value in Appendix B to Part 20 for airborne uranium ore dust in restricted areas is 75 micrograms of natural uranium per cubic meter of air* (from paragraph 4 of the Note to Appendix B)** or, if gross alpha counting of the air sample is performed, the concentration value is 1 x 10^{-10} microcuries per milliliter of air. This value applies to the alpha emissions of uranium-238, uranium-235 (negligible), uranium-234, thorium-230, and radium-226. If chemical separation and counting or spectrometric procedures are used to determine the alpha emission rate from uranium alone, the concentration value is 5 x 10^{-11} microcuries of uranium per milliliter of air. The uranium ore dust concentration value is applicable to areas where ore dust is handled prior to chemical separation of the uranium from the ore. Where the ore crushing and grinding circuits, chemical leaching areas, and yellowcake areas are physically isolated from each other, the ore dust value obviously applies to the ore handling areas.

Where ore handling and yellowcake processing are not physically isolated from each other, it is necessary to have some knowledge of the source of the airborne material to determine whether the uranium ore dust limit applies. For example, in a mill that produces little ore dust because of a wet ore grinding process but has significant emissions from yellowcake processing equipment, the natural uranium concentration of 200 micrograms of natural uranium per cubic meter of air (or 1 x 10^{-10} microcuries per milliliter) may be applicable throughout the plant. In order to determine if uranium ore dust concentrations are sufficiently low to allow use of this limit for natural uranium, paragraph 5 of the Note to Appendix B to Part 20 must be consulted. If uranium ore dust is present in quantities below 7.5 micrograms of uranium per cubic meter of air (i.e., below 10% of the ore dust limit), uranium ore dust may be considered to be not present, and the value for natural uranium of 200 micrograms per cubic meter may be used instead of the uranium ore dust limit of 75 micrograms per cubic meter.

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*Uranium concentrations are expressed preferentially in mass units instead of radiological units because (1) uranium concentrations are usually measured in terms of mass, and (2) the intake limit is usually based on the mass of uranium in the kidney to avoid effects of chemical toxicity.

**Micrograms of uranium can be converted to microcuries by using the specific activity of natural uranium: 6.77 x 10^{-7} microcuries per microgram.

An acceptable sampling program for airborne uranium ore dust includes monthly grab samples of 60-minutes duration in worker-occupied areas in which ore is actively handled. The quantity of air sampled and the method of analysis should allow a lower limit of detection (LLD) of at least 7.5 micrograms per cubic meter or 5×10^{-12} microcuries of natural uranium per milliliter of air. However, if any area has been designated as an "airborne radioactivity area," as defined in paragraph 20.203(d), sampling frequency should be increased to weekly and, in addition, an investigation of the cause of the high levels should be made. Where practicable, engineering controls should be adopted to prevent ore handling areas from becoming airborne radioactivity areas. Outdoor areas such as the ore pad should be sampled quarterly. Sample analysis should usually be completed within two working days after sample collection. Unusual results should be reported promptly to the Radiation Safety Officer (RSO).*

Only ore dust samples representative of the air inhaled by the workers present are acceptable to the NRC staff. Samples taken between the source and the worker are considered representative. Samples should be taken while normal ore handling is taking place. The state of operation of major equipment during sampling should be recorded. In large rooms, several locations should be sampled.

During the first year of operation, new mills will need a much more extensive air sampling program than operating mills to determine what locations provide concentrations representative of the concentrations inhaled by workers.

1.2 Surveys for Airborne Yellowcake

Studies show that yellowcake, a mixture primarily of ammonium diuranate and uranium oxides, is composed of soluble (biological halflife in the lung is less than 50 days) and insoluble (biological halflife in the lung is greater than 50 days) components (Refs. 2, 3, and 4). For simplicity and added protection, it should generally be assumed that yellowcake is entirely soluble for compliance with weekly limits. Intake of soluble uranium must be limited on a weekly basis in addition to the quarterly intake limit for combinations of radionuclides because paragraph 20.103(a)(2), in connection with footnote 4 of Appendix B to Part 20, imposes a weekly intake limit of 9.6 milligrams for soluble uranium (i.e., equivalent to breathing at a rate of 1.2 cubic meters per hour for 40 hours in air containing 200 micrograms of natural uranium per cubic meter or 1×10^{-10} microcuries of natural uranium per milliliter of air). This special provision for soluble uranium is for protecting the kidneys from the chemical toxicity of uranium. The basis for Part 20 limits for soluble natural uranium is that the chemical toxicity limit is always more restrictive than the radiological limit.

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*The title "Radiation Safety Officer" is used by many licensees and, in this guide, will apply to the person responsible for conducting health physics survey programs; other titles are equally acceptable.

Thus, surveys for airborne yellowcake are necessary to demonstrate compliance with the weekly limit in paragraph 20.103(a). Surveys are also necessary to establish the boundaries of airborne radioactivity areas and to determine whether precautionary procedures such as process or other engineering controls, increased surveillance, limitation on working times, provisions of respiratory equipment, or other precautions should be considered in compliance with paragraph 20.103(b).

Surveys for yellowcake are more extensive than ore dust surveys; yellowcake is a greater potential inhalation hazard than uranium ore dust because of its concentrated form. The lung, bone, and kidney are the organs primarily subject to radiation from uranium. In addition, yellowcake is a chemical toxicity hazard to the kidney. If the kidney is adequately protected from yellowcake, no organ dose limits will be exceeded according to the dosimetric models forming the basis for the Commission's regulations.

The recommended survey program for yellowcake uses a combination of (1) area air sampling and (2) breathing zone sampling during operations that may involve considerable intake such as those that require a special work permit.

Area air sampling for yellowcake should be performed weekly in airborne radioactivity areas and monthly in areas where yellowcake is present but not in sufficient quantity for the area to be designated an airborne radioactivity area. The increased surveillance of airborne radioactivity areas should be performed to meet the requirement in paragraph 20.103(b)(2) for such areas.

Samples should be representative of the air inhaled by the workers. The state of operation of major equipment during sampling should be noted. The quantity of air sampled and the method of analysis should allow an LLD of at least 20 micrograms of uranium per cubic meter of air (10% of the Appendix B concentration). Breathing zone sampling for specific jobs should be used to monitor intakes of individual workers doing jobs that are likely to involve more than 10 MPC-hours* in any one job. An example of a job during which such breathing zone sampling may be used is maintenance of yellowcake drying and packaging equipment. Sample analysis should usually be completed within one working day after sample collection. Unusual results should be reported promptly to the RSO, and an investigation of the causes should be made.

1.3 Surveys for Radon-222 and Its Daughters

In uranium mills, significant concentrations in air of radon and its daughters may occur near ore storage bins and crushing and grinding circuits or in enclosed locations where large quantities of ore are found, particularly dry ore.

NRC regulations permit measurements of concentrations of either radon-222 itself or the radon daughters. However, at uranium mills, measurements of daughters are considered by the staff to be more appropriate. Radon daughter measurements are both easier to obtain and more relevant in terms of hazard to the worker.

Monthly measurements of radon-222 daughter concentrations should be made where radon-222 daughters routinely exceed 10% of the limit or 0.03 working level (i.e., the daughter concentrations are present according to paragraph 5 of the Note to Appendix B to Part 20). If radon-222 daughter concentrations are normally greater than 0.08 working level (25% of limit), the sampling frequency should be increased to weekly. Quarterly sampling for radon-222 daughters should be made where previous measurements have shown the daughters are not generally present in concentrations exceeding 0.03 working level (10% of the limit) but where proximity to sources of radon-222 daughters might allow them to be present. For example, quarterly measurements might be appropriate for a shop area attached to the crushing and grinding circuit building.

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*MPC is the acronym for maximum permissible concentration.

Radon-222 daughter samples should be taken so as to be representative of worker exposures. Samples should be taken near locations where workers are most often present. The state of operation of major equipment during sampling and the time of day the sample was taken should be recorded.

The lower limit of detection for radon-222 daughter measurements should be 0.03 working level so that concentrations defined as being present in paragraph 5 of the Note to Appendix B to Part 20 can be detected. The lower limit of detection (LLD) for a single measurement is defined as:

$$LLD = 4.66S$$

where S is the standard deviation of the normal background, i.e., alpha emitters on a filter after radon daughters have decayed away.*

However, measured values less than the lower limit of detection, including negative values, should still be recorded on data sheets. The lower limit of detection is set high enough to provide a high degree of confidence that any measured value above the LLD truly represents radon daughters and is not a "false positive" value. However, the most accurate annual average for an area is still obtained by averaging all measured values, including values obtained that are below the lower limit of detection.

The modified Kusnetz method for measuring radon daughter working levels is a suitable method for uranium mills. The procedure consists of sampling radon daughters on a high efficiency filter paper for 5 minutes and, after a delay of 40 to 90 minutes, measuring the alpha activity on the filter. The original Kusnetz method used a rate meter to measure alpha activity. In the modified Kusnetz method, the rate meter is replaced by a scaler. This improves the sensitivity to a practical lower limit of 0.03 working level for a 1-minute count on a 10-liter (0.01 cubic meter) sample. This is about a factor of 10 lower than that originally obtained using the original Kusnetz method. A 4-minute count gives a lower limit of about 0.003 working level (Ref. 1).

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*This definition of LLD was chosen to be consistent with the NRC position previously stated in Tables 1 and 3 of Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants." The basis for the definition is given in Reference 5. The definition is also used in other Regulatory Guides, among them 4.14, "Radiological Effluent and Environmental Monitoring at Uranium Mills," and 8.14, "Personnel Neutron Dosimeters."

High efficiency membrane or glass fiber filters should be used to minimize loss of alpha counts by absorption in the filter. However, a corrective factor to account for alpha absorption in the filter paper should still be used. Care should be taken to avoid contamination of the alpha counter.

The modified Kusnetz method is discussed in more detail in References 1 and 6. Other acceptable methods discussed in Reference 1 are the original Kusnetz method with greater than 10 liters of air sampled, the modified Tsivoglou method, and the Rolle method. The modified Tsivoglou method is slightly more accurate but is also more complicated than the modified Kusnetz method. The Rolle method is quicker than the Kusnetz method, but is less sensitive. Alpha spectroscopy yields acceptable results, but the instruments are expensive and fragile and lack portability. Recently "instant working level" meters have been developed, which have the advantage of speed. These are also acceptable if an LLD of 0.03 working level can be achieved.

1.4 Surveys for External Radiation and Action Levels for External Radiation

Most, but not all, mill workers receive external gamma radiation doses of less than 1 rem per year (Ref. 1). Gamma radiation exposure rates are generally below 1 milliroentgen per hour in contact with incoming ore and are about 1.2 milliroentgens per hour in contact with fresh yellowcake (Ref. 1). During the buildup of the uranium daughters thorium-234 and protactinium-234 in fresh yellowcake, the radiation levels increase somewhat for several months following yellowcake production.

Gamma radiation surveys should be performed semiannually throughout the mill at locations representative of where workers are exposed in order to allow determination of radiation area boundaries in accordance with paragraph 20.203(b) and to determine external radiation dosimetry requirements, in accordance with § 20.202. At new mills a gamma radiation survey should be performed shortly after plant operation starts.

If the semiannual survey reveals any areas accessible to personnel where the gamma exposure rates are high enough that a major portion of the body of an individual could receive a dose in excess of 5 millirems in any hour or a dose in excess of 100 millirems in any 5 consecutive days, the area must be designated a "radiation area," as defined in paragraph 20.202(b)(2), and surveys of this area should be made quarterly. For example, if the maximum time any individual worker spends in a room in a 5-day period is 40 hours, the room will be a radiation area if the exposure rate exceeds 2.5 milliroentgens per hour. Surveys performed for this purpose should be representative of where workers sometimes stand so that their whole-body radiation exposures can be estimated. Thus, measurements should generally be made at about waist height and 12 inches from surfaces (see § 20.204(a) and Item 6(a) of Regulatory Guide 10.6, "Guide for the Preparation of Applications for Use of Sealed Sources and Devices for the Performance of Industrial Radiography"). Use of surface "contact" exposure rate measurements are not required for establishing radiation area boundaries or estimating personnel whole-body exposures because these exposures would not be representative of the exposures workers would receive.

The existence of a radiation area in a mill should be considered an action level. If a radiation area exists, the cause of the elevated gamma radiation levels should be investigated, and actions to reduce the radiation levels should be taken if practicable.

A list of the radiation levels in each area of the plant should be prepared after each survey. The number of areas on the list should be held to a manageable number. In general, a minimum of 20 survey locations is necessary to characterize the radiation levels in the mill.

To determine the need for personnel monitoring, quarterly radiation exposures expected for each category of plant worker should be calculated from the measured radiation levels and predicted occupancy times. If the calculated quarterly radiation gamma ray exposure for any individual worker exceeds 0.31 rem, § 20.202 of 10 CFR Part 20 requires that the worker wear a personnel radiation dosimeter (e.g., film badge or TLD). In addition, personnel monitoring should be used for at least a 1-year period to verify the survey results even if predicted levels are below 0.31 rem. If external radiation doses to any worker exceed 0.31 rem per quarter, an investigation of the causes should be made and corrective actions taken if appropriate.

In addition to gamma surveys, beta surveys should be made every two years to estimate extremity and skin exposures for workers who work for long periods in close proximity to yellowcake. This survey does not require more frequent measurements of yellowcake beta levels since those beta levels are fairly constant. The beta dose rate from a uranium metal slab in which equilibrium with protactinium-234 and thorium-234 has been reached is about 240 millirems per hour. The dose level from fresh yellowcake will be below this because of lower uranium density, self-shielding by oxygen and other elements, and lower concentrations of protactinium-234 and thorium-234 that have not regained equilibrium. In addition, rubber gloves will largely eliminate the beta dose to the hands if used. Extremity monitoring is required by paragraph 20.202(a) for any worker whose hand dose would exceed 4.68 rems in a quarter.

It should be noted that commercially available film badge and TLD services often cannot measure beta radiation in the mixed beta-gamma field of a uranium mill (see, for example, Tables A-11 and A-12 of Reference 7). Workers' beta doses should be estimated from survey instrument measurements rather than from such dosimetry data.

1.5 Surveys for Surface Contamination

NRC regulations provide no specific limit on surface contamination levels in restricted areas. However, yellowcake or ore dust lying on surfaces can become resuspended in the air if they are disturbed. Thus, surface contamination presents a potential source of hazardous airborne material; it can become resuspended and contribute to the intake of radionuclides, which is limited by paragraph 20.103(a).

In ore-handling areas, surface contamination is not a serious problem because of the very low specific activity of the ore. In fact, cleanup attempts by methods such as sweeping are likely to produce a more serious hazard through resuspension in the air than if the ore dust were allowed to remain where it lies. When necessary, cleanup may be performed by hosing down the ore dust into floor sumps or by using vacuum suction systems.

In leaching and chemical separation areas there is usually little dust and little difficulty with surface contamination during normal operations.

In the precipitation circuit and the yellowcake drying and barrelling areas, surface contamination can be a problem because of the concentrated nature of the yellowcake. The IAEA recommends (in Reference 1) a limit for surface contamination on walls, floors, benches, clothing, etc., of 10^{-3} μCi/cm^2 (220,000 dpm/100 cm^2), which is equivalent to almost 2 mg/cm^2 of natural uranium. Based on experience, the IAEA concluded that if surface contamination levels are kept below this value, the contribution to airborne radioactivity from surface contamination will be well below applicable limits. The British National Radiological Protection Board also recommends a limit of 10^{-3} $\mu C/cm^2$ for uranium contamination in active areas of plants (Ref. 8), based on calculations using resuspension factors rather than experience. The NRC staff considers these levels acceptable to meet the ALARA concept in uranium mills. The levels are low enough to ensure little contribution to airborne radioactivity, yet not so low that the cost of meeting them in uranium mills is too high. Such an amount of surface contamination is readily visible because of the low specific activity of uranium and does not require a survey instrument for detection. Daily visual inspections should be made for locating yellowcake contamination on surfaces.

Visible yellowcake should be cleaned up promptly, especially where contamination will be disturbed and resuspended on walkways, railings, tools, vibrating machinery, and similar surfaces. Spills should be cleaned up before the yellowcake dries so that resuspension will be lessened. Surfaces on which yellowcake may settle should be painted in contrasting colors to make the yellowcake more visible.

In rooms where work with uranium is not performed, such as eating rooms, change rooms, control rooms, and offices, a lower level of surface contamination should be maintained. These areas should be spot-checked weekly for removable surface contamination using smear tests and for total contamination using instrument measurements. The areas should be promptly cleaned if surface contamination levels exceed 25% of the values in Table 1.

TABLE 1

SURFACE CONTAMINATION LEVELS
FOR URANIUM AND ASSOCIATED DECAY PRODUCTS
(REFERENCES 9 AND 10)

Average (total)	5,000 dpm alpha per 100 cm^2	Averaged over no more than 1 m^2
Maximum (total)	15,000 dpm alpha per 100 cm^2	Applies to an area of not more than 100 cm^2
Removable	1,000 dpm alpha per 100 cm^2	Determined by smearing with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the smear

1.6 Surveys for Contamination of Skin and Personal Clothing

Contamination of skin and personal clothing should be controlled to prevent the spread of contamination to unrestricted areas (i.e., the workers' cars and homes). Alpha radiation from uranium on the skin or clothing is not a direct radiation hazard because the alpha particles do not penetrate the dead layer of the skin. Rather, uranium is primarily a hazard if it is inhaled or swallowed.

Visual examination for yellowcake is not sufficient evidence that the worker's skin or clothing is sufficiently free of contamination to permit the workers to leave the work environment. Normally such contamination can be adequately controlled if workers wash their hands before eating, shower before going home, and do not wear street clothes while working with yellowcake in the mill. Prior to leaving the restricted area, all mill employees should either shower or monitor themselves after changing clothes. An alpha survey instrument should be available at the exit of the employee change room. If clothing is not changed, clothing should be monitored. In addition, the licensee should perform at least quarterly using an alpha survey instrument an unannounced spot survey for alpha contamination on workers leaving the mill. The licensee should adopt an action level of 1000 dpm per 100 square centimeters. If levels exceed this action level, additional showering or decontamination should be done, and an investigation of the cause of the contamination should be made. These levels are considered by the NRC staff to meet the ALARA concept in a uranium mill.

1.7 Surveys of Equipment Prior to Release to Unrestricted Areas.

Surface contamination surveys should be conducted before potentially contaminated equipment is released to unrestricted areas. If contamination is detected, the equipment should be decontaminated until additional efforts do not significantly reduce contamination levels. The applicant may propose and justify total and removable contamination limits below which release of the equipment to unrestricted areas is permitted. The surface contamination levels listed in Table 1 are acceptable to the licensing staff and need not be justified by the applicant (Refs. 9 and 10).

1.8 Survey of Packages Prepared for Shipment

Smear tests of external surfaces of yellowcake packages prepared for shipment should be carried out after packaging. Packages having contamination should be cleaned until the packages meet Department of Transportation requirements for nonexclusive-use vehicles in § 173.397 of 49 CFR Part 173; i.e., removable alpha contamination must be below 22,000 dpm per 100 cm^2 averaged over 300 cm^2.

1.9 Ventilation Surveys

Surveys of the adequacy of ventilation systems for both enclosed and unenclosed areas are discussed in Draft Regulatory Guide OH 941-4, "Information Relevant to Ensuring that Occupational Radiation Exposures at Uranium Mills Will Be As Low As Is Reasonably Achievable."

2. INTAKE AND EXPOSURE CALCULATIONS

Uranium Ore Dust and Yellowcake.

In 10 CFR Part 20, paragraph 20.103(a)(1) establishes a quarterly limit on exposure to airborne uranium ore dust, and paragraph 20.103(a)(2) establishes a weekly limit on exposure to soluble uranium.

The intake of uranium ore dust or yellowcake during the weekly or quarterly period being evaluated should be estimated using the following equation:

$$I = b \sum_{i=1}^{n} \chi_i \, t_i \qquad (1)$$

where

I = uranium intake, μg or μCi

t_i = time of exposure to average concentration χ_i, hrs

χ_i = average concentration of uranium in breathing zone air during the time t_i, $\mu g/m^3$ or $\mu Ci/m^3$

b = breathing rate, 1.2 m^3/hr

n = number of exposure periods during the week or quarter

Radon Daughters

In 10 CFR Part 20, paragraph 20.103(a)(1) establishes an annual limit on exposure to radon daughters. This annual exposure should be estimated using the following equation:

$$E = \sum_{i=1}^{n} W_i t_i \qquad (2)$$

where

E = annual radon daughter exposure, working-level months

t_i = time of exposure to W_i, months

W_i = average number of working levels in breathing zone air during the time t_i

n = number of exposure periods during the year

If the licensee's respiratory protection program is being conducted in full conformance with Regulatory Guide 8.15 and the appropriate NRC regional office has been notified that the licensee plans to use respirators, χ_i and W_i values may be divided by the prescribed protection factor in the calculation of I and E.

The preferred method for determining values of t_i is that of actual timing and recording for each exposure. However, t_i values may be derived from a time study of worker occupancy in the various mill areas provided that the elapsed time between the completion of the time study and the intake determination does not exceed 3 months. When nonroutine maintenance or cleanup operations are performed, accurate time records should be kept, and the results of special area or breathing zone samples taken over this period should be used in calculating employee exposures.

3. REPORTS OF OVEREXPOSURES TO AIRBORNE MATERIALS

Overexposure reports to the appropriate NRC regional office are required by § 20.405 if the intake of uranium ore dust or yellowcake exceeds the quantities specified in § 20.103 or if the exposure to radon and its daughters exceeds the working-level values specified in footnote 3 to Appendix B to 10 CFR Part 20. Many uranium mill workers are exposed to a combination of these materials; in such cases, Appendix B to 10 CFR Part 20 specifies the method for determining whether NRC exposure limits have been exceeded. Overexposure reports are also required for combined exposures that exceed NRC limits.

A listing of reportable overexposures for various uranium products follows:

1. Uranium Ore Dust. If during a period of 1 calendar quarter a worker is subjected to an intake of uranium in ore dust greater than 47,250 micrograms (0.03 microcuries)*, a reportable overexposure has occurred.

2. Yellowcake. If during a period of 1 calendar week a worker is subjected to an intake of uranium in yellowcake greater than 9,600 micrograms (0.0048 microcurie) or if during a period of 1 calendar quarter a worker is subjected to an intake of uranium in yellowcake greater than 126,000 micrograms (0.063 microcurie), a reportable overexposure has occurred.

3. Radon Daughters. If during a period of 1 calendar year a worker is subjected to a radon daughter exposure, caused by radon emanation from uranium ore, of more than 4 working-level months, a reportable overexposure has occurred.

4. Combination Exposures, Weekly Determination. For a worker exposed to yellowcake, it is necessary at the end of each week to determine whether a reportable overexposure has occurred. Since the 9600-microgram limit is intended to protect the kidney against chemical damage and since such damage from radon daughters is negligible, radon daughter exposure need not be considered in the weekly overexposure determination. However, in vitro solubility studies of uranium ore dust have indicated that uranium inhaled in this form will reach the kidney after buildup in the body and therefore should not be neglected in the weekly overexposure determination (Ref. 2). Therefore, the following procedure should be followed. In order to obtain the weekly average, use the ore dust uranium intake in micrograms for the most recent calendar quarter and divide by 13. This average is added to the yellowcake uranium intake in micrograms for the week in question. If the sum exceeds 9600 micrograms, a reportable overexposure has occurred.

5. Combined Exposures, Quarterly Determination. For a worker exposed to any mixture of these materials, at the end of each calendar quarter it is necessary to determine whether a reportable overexposure has occurred. The following

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*I.e., 75 micrograms/m^3 x 630 m^3 = 47,250 micrograms. Similarly, 0.5 x 10^{-10} microcuries/ml x 10^6 ml/m^3 x 630 m^3 = 0.0315 microcurie.

procedure should be used for this purpose. Any ore dust uranium intake for the quarter, in micrograms, is divided by 47,250 micrograms (0.03 microcurie) to obtain the fraction of the limit that has been experienced. The yellowcake intake for the quarter, in micrograms, is divided by 126,000 micrograms (0.063 microcurie)* to obtain the fraction of that limit that has been experienced. The number of working-level months (radon daughter exposure) for the past 4 quarters is divided by 4. If the sum of these three fractions exceeds unity, a reportable overexposure has occurred.

4. ACTION LEVELS

4.1 The 40-Hour Control Measure

The 40-hour control measure specified in § 20.103(b)(2) is an action level of concern to the uranium mill operator only in the case of exposure to uranium ore dust and radon daughters. If during a period of 1 calendar week a worker is subjected to an intake of uranium in ore dust greater than 3,600 micrograms, radon daughters exceeding 1/13 working-level month, or fractional combinations exceeding one, § 20.103(b)(2) requires that the cause must be determined, corrective action to prevent another such occurrence must be taken, and a record of the corrective action must be maintained.

4.2 Administrative Action Levels

In addition, the licensee should establish administrative action levels for exposure to airborne radioactivity in accordance with the ALARA philosophy. These action levels should be some fraction of the allowable intakes or concentrations in 10 CFR 20 and should be established for at least the following:

1. Time-weighted exposures averaged over a week
2. Concentration of airborne radioactivity for any air sample.

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*I.e., 200 micrograms/m^3 x 630 m^3 = 126,000 micrograms. Similarly, 10^{-10} microcuries/ml x 6.3 x 10^8 ml = 0.063 microcurie.

If any of these action levels are reached, an investigation should be performed by the RSO and any necessary action should be taken to ensure that both exposures and airborne concentrations are maintained as low as is reasonably achievable.

5. ESTABLISHMENT OF "AIRBORNE RADIOACTIVITY AREAS"

In general, yellowcake drying and packaging rooms and enclosures should always be considered to be airborne radioactivity areas because of the high concentrations that can result if any equipment malfunctions. On the other hand, ore crushing and grinding areas and areas outside yellowcake drying and packaging areas will not normally need to be classified as airborne radioactive areas when normal engineering controls are used. Any area, room, or enclosure is an "airborne radioactivity area" as defined in paragraph 20.203(d) if (1) at any time the uranium ore dust concentration or yellowcake concentration averaged over the different sampling locations exceeds 75 micrograms per cubic meter or 200 micrograms per cubic meter of air, respectively (i.e., the values in Appendix B to 10 CFR Part 20), or (2) the concentration exceeds 25 percent of the values in Appendix B to 10 CFR Part 20 averaged over the number of hours in any one week in which individuals are present in such area, room, or enclosure. For example, an area that is occupied 20 hours per week (out of the 40 hours assumed as a basis for the limits), is an airborne radioactivity area if the concentration of uranium in yellowcake exceeds 100 micrograms per cubic meter of air. The licensee must, of course, maintain adequate records to show that occupancy was in fact thus limited. If radon-222 daughters or ore dust are also present (see Section C.1.3 of this guide), their concentrations divided by the Table I Appendix B value should be added. If the sum exceeds unity or exceeds 0.25 after adjustment of the Appendix B value for the occupancy factor, the area is an airborne radioactivity area.

6. POSTING OF CAUTION SIGNS, LABELS, AND NOTICES TO EMPLOYEES; ACCESS CONTROLS

The radiation protection staff should periodically survey to ensure that signs, labels, required notices to employees, copies of licenses, and other items are properly posted as required by 10 CFR Part 20.

The mill and tailings area should be fenced to restrict access, and the fence should be posted with "Caution, Radioactive Material" signs as required in paragraph 20.203(e)(2). If the fence and all entrances are thus posted and in addition contain the words "Any area within this mill may contain radioactive material," the entire area is posted adequately to meet the requirement in paragraph 20.203(e)(2). Additional posting of each room with "Radioactive Material" signs is not necessary. "Radiation Area" and "Airborne Radioactivity Area" must still be posted in accordance with paragraphs 20.203(b) and (d). Parking for employees and visitors should be outside the restricted area.

7. CALIBRATION OF SURVEY INSTRUMENTS

Portable survey instruments should be placed on a routine maintenance and calibration program to ensure that properly calibrated and operable survey instruments are available at all times for use by the health physics staff.

An adequate calibration of survey instruments cannot be performed solely with built-in check sources. Electronic calibrations that do not involve a source of radiation will not determine the proper functioning and response of all components of an instrument. However, an initial calibration with a radiation source and periodic tests using electronic input signals may be considered adequate for high ranges of survey instruments that are not used routinely.

Survey instruments should be checked prior to each usage. This check should be supplemented at 6-month intervals or at the manufacturer's suggested interval, whichever is shorter, with a calibration of each instrument at two points separated by at least 50 percent of each linear scale routinely used or with a calibration at one point near the midpoint of each decade on logarithmic scales that are routinely used. Digital readout instruments with either manual or automatic scale switching should be calibrated in the same manner as are linear readout instruments. Digital readout instruments without scale switching should be calibrated in the same manner as are logarithmic readout instruments. Survey instruments should be calibrated following repair. A survey instrument may be considered properly calibrated when the instrument readings are within ±10 percent of the calculated or known values for each point checked. Readings within

±20 percent are considered acceptable if a calibration chart or graph is prepared and attached to the instrument.

Errors in estimates of the volume of air that has passed through filters should be avoided by accurate calibration of the flow rate and by preventing or correcting for the loss of flow caused by accumulation of material on the filter. Air flow rates through filters should be determined by calibrating pumps with the filter paper in place once every 6 months to ±20% accuracy. These calibrations should be done in accordance with manufacturer's recommendations. Further information on these calibrations is contained in Draft Regulatory Guide OH 905-4, "Calibration and Error Limits of Air Sampling Instruments for Total Volume of Air Sampled."

The fluorometric analysis system should be calibrated by running a known standard uranium and a blank with each batch.

Alpha counting systems used for radon daughter measurements should be calibrated at least monthly by using a known standard alpha source.

8. PROTECTIVE CLOTHING

Workers working in airborne radioactivity areas should be provided with protective clothing such as coveralls and shoes or shoe covers. Gloves should be used when yellowcake will be handled because of its high beta dose rate. Rubber gloves should provide almost total shielding against the beta rays.

Protective clothing should be changed and discarded or laundered weekly or whenever yellowcake is visible on the clothing. Potentially contaminated clothing should not be sent to a laundry that is not specifically authorized by the NRC or an agreement state to process clothing contaminated with uranium unless the clothing has been surveyed and found to have less uranium contamination than the values in Table 1 of this guide.

9. QUALITY ASSURANCE PROGRAM

The licensee should ensure the accuracy of survey measurements by having a quality assurance program. Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Program (Normal Operations)--Effluent Streams and the Environment," should be consulted for guidance on quality assurance.

REFERENCES

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DRAFT VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

Applicants for a uranium milling license must submit a license application containing the information specified in Regulatory Guide 3.5, "Standard Format and Content of License Applications for Uranium Mills." The purpose of this proposed action is to describe health physics surveys to protect workers that are acceptable to the NRC staff. Information about health physics surveys is covered under Section C.5, "Operations," in Regulatory Guide 3.5.

1.2 Need

Licensees are now uncertain what the NRC staff will accept in the way of a health physics survey program to protect workers. As a consequence, a wide variety of programs are submitted. In order to meet minimum acceptable standards, much correspondence between the applicant and NRC is required. A guide will reduce the amount of correspondence needed, save manpower for both NRC and the applicant, show clearly how NRC regulations apply to uranium mills, and establish a uniform standard for an acceptable survey program for worker protection.

1.3 Value/Impact

1.3.1 NRC

The impact of the proposed guidance will be primarily to reduce licensing staff effort expended in reviewing applications and corresponding with applicants in areas where the application does not meet acceptable NRC licensing standards. An estimated 0.5 man-year is required to develop the guide.

1.3.2 Other Government Agencies

The proposed guidance will impact on the Mine Safety and Health Administration (MSHA) because they also regulate occupational health protection at uranium mills and on Agreement State regulatory agencies that regulate mills, primarily agencies in New Mexico, Colorado, Texas, Washington, and Florida. A Memorandum of Understanding (MOU) signed by NRC and MSHA states that each agency will coordinate the development of standards with the other agency. The MOU was published in the Federal Register (45 FR 1315) on January 4, 1980.

1.3.3. Industry

Industry will benefit from having clear guidance on what constitutes NRC licensing policy. The total cost of the occupational health physics program (surveys plus other parts of the program) is estimated to be roughly 4 man-years per year or about $300,000 per year per mill when the costs of overhead, supplies, equipment, and contracted services are included. This does not include the cost of the environmental and effluent monitoring program nor does it include amortization costs on equipment in the mill installed to limit occupational exposure. Equipment design is not covered in this guide, therefore, costs are not estimated here. However, the annual amortization and operating costs of equipment installed to protect workers is not negligible.

1.3.4 Workers

Workers' protection should improve from having clearly stated and consistent standards for health physics survey programs. Workers and workers' representatives will now have access to a clearly defined standard health physics survey program. This will help them understand whether their employer has an adequate program and why some things are done as they are.

1.3.5 Public

The guidance pertains to worker protection programs. It will not directly affect the public.

1.4 Decision

The NRC should develop guidance on standard health physics survey programs for worker protection that are acceptable to the NRC licensing staff.

2. TECHNICAL APPROACH

The technical approach in the guidance will be based on (1) NRC licensing policy as expressed in Safety Evaluation Reports (SER) written by the NRC licensing staff, especially the recent SER for Minerals Exploration Company Sweetwater

Uranium Project; (2) the IAEA Manual on Radiological Safety in Uranium and Thorium Mines and Mills, IAEA Safety Series No. 43, 1976; and (3) other references to be cited in the guidance.

3. PROCEDURAL APPROACH

3.1 Procedural Alternatives

There are three reasonable procedural alternatives: (1) publish the guidance directly in the regulations, (2) publish the guidance in a regulatory guide, and (3) continue to provide guidance to applicants on a case-by-case basis.

3.2 Value/Impact of Procedural Alternatives

A regulation is not a really suitable vehicle for the type of guidance envisioned because much of a health physics survey program must be tailored to the design and needs of the individual mill. Regulations do not provide for the needed flexibility.

A Safety Evaluation Report is not the most desirable way to distribute the guidance because these reports are not widely circulated to the industry and public, and each Safety Evaluation Report differs from others because of the needs and desires of individual mill licensees.

The guidance envisioned seems best suited for a regulatory guide. A guide provides the best mix of flexibility and clear statement of a uniform and consistent licensing policy.

3.3 Decision on Procedural Approach

Developing a regulatory guide is the favored procedural approach.

4. CONCLUSION

In summary, it is proposed that a regulatory guide on health physics survey programs for worker protection in uranium mills be developed.

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NUCLEAR REGULATORY COMMISSION
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DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

September 1980
Division 1
Task SS 926-4

Contact: L. Brown (301) 443-5976

PROPOSED REVISION 1* TO REGULATORY GUIDE 1.23

METEOROLOGICAL PROGRAMS IN SUPPORT OF NUCLEAR POWER PLANTS

A. INTRODUCTION

Paragraph 100.10(c)(2) of 10 CFR Part 100, "Reactor Site Criteria," states that, in determining the acceptability of a site for a power or test reactor, the Nuclear Regulatory Commission (NRC) will take into consideration meteorological conditions at the site and in the surrounding area.

Paragraph 50.36a(a)(2) of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires nuclear power plant licensees to submit semiannual reports specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and gaseous effluents and such other information as may be required by the NRC to estimate maximum potential annual radiation doses to the public resulting from effluent releases. A knowledge of meteorological conditions in the vicinity of the reactor is important in providing a basis for estimating annual radiation doses resulting from radioactive materials released in gaseous effluents.

In order for the NRC to fulfill its responsibilities under the National Environmental Policy Act of 1969 and in accordance with the requirements of 10 CFR Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection," and Appendix I, "Numerical Guides for Design Objectives

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*This revised guide sets forth essentially the same considerations for preoperational programs for new facilities as the previous edition of this guide though with reorganization of content and clarification and additional details in the discussions. The major change relates to the addition of special considerations for emergency preparedness (regulatory position 8). The substantial number of changes in this proposed revision has made it impractical to indicate the changes with lines in the margin.

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by NOV 20 1980

Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

FOR COMMENT

and Limiting Conditions for Operation To Meet the Criterion 'As Low As Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," to 10 CFR Part 50, basic meteorological information must be available for use in assessing potentially adverse environmental effects of a radiological and nonradiological nature resulting from the construction or operation of a nuclear power plant.

In addition to the requirements for determining meteorological conditions at nuclear power plants in order to assess siting, licensing, and environmental factors, detailed meteorological information is necessary for dealing with radiological emergencies. Appendix E, "Emergency Plans for Production and Utilization Facilities," to 10 CFR Part 50 requires each applicant for an operating license to include in its final safety analysis report, required by paragraph 50.34(b) of 10 CFR Part 50, plans for coping with radiological emergencies. The plans must include criteria for determining when protective measures should be considered within and outside the site boundary to protect health and safety and prevent damage to property. In this regard, it is necessary for the applicant or licensee to establish and maintain a meteorological program capable of rapidly assessing critical meteorological parameters.

Thus, at each nuclear power plant site, there are multiple needs for programs that will adequately measure and document basic meteorological data. These data can be used to develop atmospheric diffusion parameters that, with an appropriate dispersion model, can be used to estimate potential radiation doses to the public resulting from actual routine or accidental releases of radioactive materials to the atmosphere or to evaluate the potential dose to the public as a result of hypothetical reactor accidents. This regulatory guide describes meteorological measurement programs acceptable to the NRC staff for providing meteorological data needed to estimate these potential radiation doses.

B. DISCUSSION

Meteorological measurement programs at a nuclear power plant site should be capable of providing the meteorological information required to make the following assessments:

1. A conservative assessment by the applicant or licensee and the NRC staff of the potential dispersion of radioactive material from and the radiological consequences of design basis accidents to aid in evaluating the

acceptability of a site and the adequacy of engineered safety features for a nuclear power plant in accordance with the requirements of 10 CFR Part 100.

2. An assessment by the applicant or licensee and the NRC staff of the maximum potential annual radiation dose to the public resulting from the routine release of radioactive materials in gaseous effluents. These assessments assist in demonstrating that operations will be or are being conducted within the limits of 10 CFR Part 20, "Standards for Protection Against Radiation," and in ensuring that effluent control equipment design objectives and proposed operating procedures meet the requirements of Appendix I to 10 CFR Part 50.

3. A realistic assessment by the applicant or licensee and the NRC staff of the potential dispersion of radioactive materials from and the radiological consequences of a spectrum of accidents to aid in evaluating the environmental risk posed by a nuclear power plant in accordance with 10 CFR Part 51.

4. A realistic assessment by the applicant or licensee and the NRC staff of such nonradiological environmental effects as fogging, icing, and salt drift from cooling towers to aid in evaluating the environmental impact of a nuclear power plant in accordance with 10 CFR Part 51.

5. An assessment by the licensee and other appropriate persons of the radiological consequences of an accidental release of radioactive material to the atmosphere. The assessments should be used to provide guidance to persons assigned to the licensee's emergency organization and to appropriate local, State, and Federal agencies with responsibilities for coping with emergencies for use in determining (a) the need for notification and participation of local and State agencies and the NRC and other Federal agencies and (b) when appropriate measures should be taken to protect public health and safety and prevent damage to property in accordance with Appendix E to 10 CFR Part 50. Meteorological measurement programs should provide an adequate basis for short distance (less than 16 km) atmospheric dispersion calculations. Other regional meteorological data are necessary to make dispersion estimates for long distances. To ensure that the required data are readily available, it is important that the applicant or licensee establish and maintain contact with the Meteorologist-In-Charge at appropriate National Weather Service Offices[1] and inventory and

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[1] The name and address of the Meteorologist-In-Charge may be obtained by contacting the Chief, Meteorological Services Division, National Weather Service, National Oceanic and Atmospheric Administration, Silver Spring, Maryland 20910.

make available to emergency response organizations and appropriate local, State, and Federal agencies meteorological data from other well-maintained meteorological systems in the plant vicinity.

Specific guidance concerning the dispersion models to be used for evaluating the potential radiological consequences of design basis reactor accidents is given in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." Guidance concerning use of site-specific meteorological information by the NRC staff is being developed for probabilistic assessment of consequences of a spectrum of accidents to aid in evaluating risks of operation of the nuclear power plant consistent with the requirements of 10 CFR Part 51. Guidance concerning the dispersion models to be used for evaluating the potential effects of routine releases of radioactive effluents into the atmosphere is given in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors." Additional guidance concerning the dispersion models to be used for evaluating the potential radiological consequences for incident response is given in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants."

When establishing a meteorological program for an initial site survey, it is essential that care be taken to locate the stations at positions where the measurements will accurately represent the overall site vicinity meteorology and, if possible, where wind patterns will not be significantly influenced by singular topographic features or by construction of plant structures at a later date.

The number of locations in the site vicinity at which meteorological measurements are necessary will depend largely on the complexity of the terrain in the vicinity of the site. For example, a site in a valley or a site near a large body of water may require multiple measuring points to determine airflow patterns and spatial variations of atmospheric stability.

C. REGULATORY POSITION

This section describes suitable meteorological measurement programs to provide the data needed to determine meteorological conditions in the vicinity of nuclear power plants in order to assess safety and environmental factors prior to plant operation and the data needed to determine when measures should be considered to protect health, safety, and property during the operational phase of the nuclear power plant.

1. METEOROLOGICAL PARAMETERS

To obtain the basic meteorological information required for estimates of atmospheric transport and diffusion and plant impact on the environment at a particular site, instrumentation that is capable of measuring wind direction and wind speed at a minimum of two levels and air temperature difference between a minimum of two levels should be provided on one tower or mast. Precipitation should be measured at or near this tower. Instrumentation should be provided for measuring air temperature on at least one level of the tower or mast corresponding to at least the measurement height of the lower level of the primary air temperature difference measurement. Instrumentation should be provided for measuring ambient moisture (relative humidity, dew point, or wet bulb temperature) on at least one level of the tower or mast. Visibility and solar radiation measurements may be necessary in conjunction with cooling system assessments.

Temperature difference with height may be used to define both the horizontal and vertical standard deviations of material distribution (σ_y and σ_z). The classification of stability parameters that define these standard deviations are delineated in Table 1. Other estimators of stability parameters such as the standard deviation of vertical wind direction (σ_ϕ) in conjunction with minimum wind speed criteria, the standard deviation of horizontal wind direction (σ_θ) in conjunction with minimum wind speed criteria, or Richardson Number may be considered (Ref. 1). However, use of and classification by alternative estimators other than temperature difference with height should be justified and may also require modification of the models described in Regulatory Guides 1.111 and 1.145 with appropriate justification.

The use of σ_θ values is not restricted to that of an indicator of diffusion. During plant operation and in the context of real-time diffusion assessments for emergency conditions, the wind direction variability is an essential element in describing the extent of the plume exposure pathway and estimating potential radiological doses.

Sites with unusual air quality characteristics (e.g., high concentrations of airborne particulates due to ambient conditions or plant operation) may require additional instrumentation (e.g., atmospheric sampling equipment). Additional wind, temperature, humidity, and precipitation instrumentation may be required to document site vicinity meteorological conditions due to complex mesoscale features (e.g., nonuniform terrain, coastal effects).

A particular site may warrant use of special meteorological instrumentation, data analysis techniques, or field studies. Proposed activities of this nature and the rationale for performing them should be discussed with the NRC staff prior to installation of special meteorological instrumentation or performance of special studies.

For making estimates of atmospheric transport and diffusion to a distance of 80 kilometers (50 mi) from the plant site, additional information may be needed. If so, it may be obtained, at least in part, from stations with well-maintained meteorological systems (e.g., National Weather Service, military stations, and any other micro-meteorological stations) if these existing stations are in locations that will aid in the description of regional airflow patterns.

2. SITING OF METEOROLOGICAL INSTRUMENTS

The meteorological tower site should represent as closely as possible the same meteorological characteristics as the region into which any airborne material will be released. Whenever possible, the base of the tower or mast should be sited at approximately the same elevation as the finished plant grade. The tower should be located in an area where singular natural or man-made obstructions or the heat dissipation system to be used during plant operation will have little or no influence on the meteorological measurements. The height of natural or man-made obstructions to air movement should ideally be lower than the measuring level to a horizontal distance of 10 times the measuring level height. Whenever possible, locating the tower or mast directly downwind of the obstructions or heat dissipation system under the prevailing downwind wind directions should be avoided. Instrumentation should be located on booms oriented into the prevailing wind direction at a minimum distance of two tower widths from the tower to preclude substantial influence of the tower upon the measurements (Ref. 2). The aspirated temperature shields should either be pointed downward or laterally toward the north.

On the primary tower, wind speed and direction should be monitored at approximately 10 and 60 meters and at a representative higher level for stack releases. The 10-meter level has been generally accepted throughout the world as a standard meteorological reference measurement level. The 60-meter level generally coincides with the routine release level for LWRs. Ambient temperature should be monitored at approximately 10 meters, and ambient moisture should be monitored at approximately 10 meters and also at a height where the measurements will represent the resultant atmospheric moisture content if cooling towers are to be used for heat dissipation. Temperature difference should be measured between the 10- and 60-meter levels and between the 10-meter and a higher level that is representative of diffusion conditions from stack release points.

If supplementary instrumented towers or masts are used to better define atmospheric conditions in the site vicinity, they should have locations and exposures that are indicative of meteorological conditions in the region of the plant site for which better definition is needed (e.g., emergency planning zones).

At coastal sites, the primary meteorological tower should be in such a location that the upper measuring level is within the thermal internal boundary layer during onshore flow conditions. Heights of the internal boundary layer should be confirmed experimentally before the tower or mast site is chosen. For a site with a simple coastline, a secondary tower or mast should be placed at a location where measurements representative of conditions in the unmodified marine air can be determined. When measurements are made to define the meteorological conditions in the vicinity of a nuclear power plant site with a neighboring area of land across a body of water, one secondary tower should be located in the water or two secondary towers or masts should be located on opposite shores. Instrument heights should be selected on the primary tower so that measurements representative of conditions within the internal boundary layer are obtained while the 50-meter separation between levels is maintained. On secondary towers or masts, instrument heights should meet the same criteria for the internal boundary layer over the shoreline.

At a valley site, the primary meteorological tower should be located so that the meteorological measurements are representative of conditions at the potential points of release. All levels at which measurements are made should be within the same thermal internal boundary layer. Drainage conditions and inversion depths should be confirmed experimentally before the tower site is chosen. For a site with a simple nonmeandering valley that is relatively uniform in depth, a secondary tower or mast should be placed at a location where measurements representative of meteorological conditions outside the valley can be determined. For a site with complex terrain, additional secondary towers or masts should be located to represent complex flow conditions.

3. DATA RECORDERS

For data acquisition on the primary tower, a dual recording system consisting of one digital and one auxiliary analog system should be used. Both system accuracies should be within the specifications presented in regulatory position 4. The wind speed and direction analog recorders should be of the continuous strip chart recording type. Multipoint strip chart recorders are considered to be sufficient for recording all other parameters. All digital records except precipitation should consist of data sampled at intervals no longer than 60 seconds. Precipitation should be recorded on a cumulative basis at least once per hour. The standard deviation of horizontal wind direction fluctuations, σ_θ, should be determined from no less than 180 instantaneous values of lateral wind direction during the recording period (e.g., if the record period is 15 minutes, values sampled at intervals of 5 seconds or less are acceptable; likewise, if the record period is 1 hour, sampling intervals of 20 seconds or less are acceptable).

The data from the primary meteorological system (backup system when necessary) should be displayed in the control room for use during plant operation. These data should also be displayed in the onsite technical support center and nearsite emergency operations facility as needed (e.g., emergency situations, training exercises, demonstrations). These data should include wind direction and speed and an indicator of atmospheric stability for the past 12-hour period representative of each potential release level. Fifteen minutes is the maximum acceptable averaging period for these data. This display should be easily accessible and should be labeled so that the information is clearly understood (e.g., direction from which the wind is blowing, Pasquill stability class).

4. SYSTEM ACCURACY

Parameter accuracy for a system refers to the composite accuracy reflecting the errors introduced by sensor, cable, signal conditioner, humidity, temperature environment for signal conditioning and recording, recorders, and data reduction process. The errors introduced by each of the separate components of the system should be determined by statistical methods (Ref. 3). All sensors should have appropriate accuracies to meet the digital system accuracies specified below over the range of environmental conditions expected to occur during the lifetime of the plant. For individual samples, all components from the sensors to the recording systems that contribute to measurement error are collectively defined by the root sum of the squares (RSS) method as the system accuracy. The RSS is calculated by squaring each error, summing the squared errors, and taking the square root of the sum. For time-averaged values, those parts of the error budget that are truly random may be decreased from their instantaneous value by dividing by the square root of the number of samples used to define the average value. Then the RSS calculation can be made.

a. For digital systems, specific accuracies of time-averaged values by parameter should be:

(1) Wind direction: ±5° of azimuth, with a starting threshold of less than 0.45 m/s (1 mph). If the wind direction sensor is to be used for the collection of σ_θ data, the damping ratio must be 0.4 to 0.6, inclusive, with a deflection of 15 degrees and delay distance not to exceed 2 meters.

(2) Wind speed: ±0.22 m/s (0.5 mph) for speeds less than 11.13 m/s (25 mph), with a starting threshold of less than 0.45 m/s (1 mph).

(3) Temperature: ±0.5°C (0.9°F).

(4) Temperature difference: ±0.15°C (0.27°F) per 50-meter height interval.

(5) Dew point: ±1.5°C (2.7°F) or an equivalent accuracy for relative humidity or wet bulb temperature. These accuracies are applicable for conditions where relative humidity is in excess of 60 percent and temperature is between -30° and 30°C (-22° and 86°F), which is the region of concern for evaluation.

(6) Precipitation: by a recording rain gauge with a resolution of 0.25 mm (0.01 in.). The accuracy of the recorded value must be within ±10 percent of the total accumulated catch.

(7) Time: within 5 minutes of actual time for all recording systems.

All parameters not covered above should be consistent with the current state of the art for the measurement of these parameters.

b. For analog systems, specific accuracies of time-averaged values by parameter should be the same as those above except that the accuracies for wind speed and direction records should be not more than 1.5 times those stated in regulatory position 4a. The system accuracies should include the reduction of data from the strip chart recorder to digital form.

5. INSTRUMENT MAINTENANCE, SERVICING SCHEDULES, AND DATA AVAILABILITY

The system should be protected against lightning and other severe environmental conditions (e.g., icing, blowing sand, salt deposition, air pollution) that may occur at the site. The meteorological measurement system and associated controlled environment housing system for the equipment should be connected to a power system that is supplied from redundant power sources. Meteorological instruments should be inspected and serviced at a frequency that will minimize extended periods of outage and ensure at least an annual 90 percent joint data recovery for atmospheric stability, wind speed, and wind direction at the level that represents each effluent release point. It is essential to maintain an adequate spare parts inventory to minimize extended periods of system outage. Annual data recovery for other individual parameters should be at least 90 percent for each parameter. Redundant sensors and recorders at appropriate locations may also be used to achieve the required data recovery. For the operational meteorological measurement program, the gap that could exist because of failure of the primary system may be filled by a backup system to ensure continuous data availability (see regulatory position 8 for additional guidance for emergency preparedness). The systems should be calibrated at least semiannually to ensure meeting the system accuracies presented in this guide; the calibration results should be reflected in the compiled data base. In areas with high ambient aerosol or particulate loadings in the atmosphere (e.g., sea coastal sites, deserts), calibrations should be performed on a more frequent basis as required to maintain system accuracies. Procedures and a log of inspection, maintenance, and calibrations should be maintained at the tower site as a controlled document and a permanent record to be made available for review. Any major modification of the system or environs should be documented and discussed with the NRC staff.

6. DATA REDUCTION AND COMPILATION

The basic reduced data should be averaged over a period of 1 hour. At least 15 consecutive minutes of continuous data during each hour should be used to represent a 1-hour average. Precipitation should be totaled hourly. The basic reduced data should be compiled into monthly and annual joint frequency distributions of wind speed and wind direction by atmospheric stability class. An example of a suitable format for data compilation and reporting purposes is shown in Table 2. The table may be modified for specific situations. For example, sites with a high occurrence of low or high wind speeds should include additional wind speed classes. Atmospheric stability should be classified as described in Table 1.

A listing should be prepared of hourly average measurements used in the evaluation, and a magnetic tape containing these parameters should be prepared in the format presented in Appendix A to this guide. An accounting of all hours should be made, with any missing data appropriately designated.

Minimum data requirements with respect to length of data record for various licensing actions are given in Section 2.3 of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition." Certain computational schemes used by the staff for probabilistic assessment of consequences of a spectrum of accidents to aid in evaluating risks of operation of the nuclear power plant require site-specific hour-by-hour meteorological data over a period of a whole year. To aid in assessing the impact of plant operation on the environment, joint frequency distribution data summaries of wind direction and speed, atmospheric stability class, and relative humidity that will permit the description of the frequency and extent of fogging and icing conditions caused by plant operation should be compiled.

7. QUALITY ASSURANCE

A quality assurance program that is consistent with the provisions of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 should be established for both the meteorological measurement program prior to nuclear power plant operation and for the meteorological measurement program in support of the operation of the nuclear power plant. Chapter 17 of NUREG 75-087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," and Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," provide further guidance.

8. SPECIAL CONSIDERATIONS FOR EMERGENCY PREPAREDNESS

Provisions should be made for remote interrogation of all utility-maintained meteorological systems during emergency situations. These systems should have the capability of being remotely interrogated simultaneously by the licensee, emergency response organizations, and the NRC without interruption of the data-gathering process. This capability may be acquired by the installation of a dial-up connection for an 80-column ASCII terminal via telephone lines (e.g., output format of RS-232-C in FSK). The transmission rate(s) should be compatible with receiving system(s) of the appropriate state(s) and counties and of the NRC. The system should have the capability of recalling 15-minute averages of meteorological parameters from at least the previous 12-hour period. The resolution of the data should meet the system accuracy specifications given in regulatory position 4. An example of a suitable format for the meteorological data is given in Appendix B to this guide.

All sites with operating nuclear power plants should have a viable backup system to obtain real-time local meteorological data. Such a system would provide meteorological information when the primary system is out of service, thus providing assurance that basic meteorological information is available during and immediately following an accidental airborne release. An independent system[2] (e.g., mobile meteorological equipment) should be established for obtaining measurements of wind direction and speed representative of the 10-meter level and a seven-category (classes A through G) estimator of atmospheric stability (e.g., temperature difference with height, wind direction fluctuations as categorized in Table 3). This information should be representative of the site environs and should include data from multiple locations when necessary. The backup system should provide information in a real-time mode in the event necessary parameters from the primary system are not available. Changeover from the primary system to the backup system should occur within 5 minutes.[3] Such information should be presented in place of the lost record as outlined in Appendix B. A functional backup communications link should also be established to ensure interrogation capability. NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," provides further guidance on special considerations for the meteorological program during the operational phase of the nuclear power plant.

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[2] An independent system can be a system installed and maintained by the licensee specifically for the purpose of providing redundant site-specific meteorological information. This system can be an existing system to which the licensee can access and should be capable of providing the designated continuous information representative of the site environs.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

This proposed revision has been released to encourage public participation in its development. Except in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method to be described in the active guide reflecting public comments will be used in the evaluation of all applications that are docketed after, or are in review at, issuance of this guide.

This guide will also be used in staff reviews conducted in accordance with emergency planning schedules as delineated in the amendments to the regulations published in the Federal Register on August 19, 1980. Paragraph 50.54(s) of 10 CFR establishes an implementation date of April 1, 1981, for emergency response plans to be upgraded and specifies April 1, 1982, as the date by which emergency support facilities, including meteorological systems, are expected to be fully operational.

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[3] The unavailability goals should be less than 0.001 for each individual parameter as outlined in NUREG-0696, "Functional Requirements for Emergency Response Facilities." Planned outages and preventative maintenance schedules should not be coincident for the primary and backup meteorological systems. The availability goals should be applied to unplanned outages, e.g., resulting from a lightning strike.

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TABLE 1

CLASSIFICATION OF ATMOSPHERIC STABILITY BY TEMPERATURE CHANGE WITH HEIGHT

Stability Classification	Pasquill Categories	Temperature Change with Height (°C/100 m)
Extremely unstable	A	$\Delta T/\Delta z \leq -1.9$
Moderately unstable	B	$-1.9 < \Delta T/\Delta z \leq -1.7$
Slightly unstable	C	$-1.7 < \Delta T/\Delta z \leq -1.5$
Neutral	D	$-1.5 < \Delta T/\Delta z \leq -0.5$
Slightly stable	E	$-0.5 < \Delta T/\Delta z \leq 1.5$
Moderately stable	F	$1.5 < \Delta T/\Delta z \leq 4.0$
Extremely stable	G	$4.0 < \Delta T/\Delta z$

TABLE 2

EXTREMELY STABLE (ΔT/Δz exceeds 4.0°C/100 m) PERIOD OF RECORD:

PASQUILL G

WIND SPEED (m/s) AT 10-m LEVEL

Wind Direction	.22-.50	.51-.75	.76-1.0	1.1-1.5	1.6-2.0	2.1-3.0	3.1-5.0	5.1-7.0	7.1-10.0	10.1-13.0	13.1-18.0	>18.0	Total
N													
NNE													
NE													
ENE													
E													
ESE													
SE													
SSE													
S													
SSW													
SW													
WSW													
W													
WNW													
NW													
NNW													
Total													

Number of Calms

Number of Invalid Hours

Number of Valid Hours

TABLE 3

CLASSIFICATION OF ATMOSPHERIC STABILITY BY SIGMA THETA

Stability Classification	Pasquill Categories	σ_θ* (degrees)
Extremely unstable	A	$\sigma_\theta \geq 22.5$
Moderately unstable	B	$22.5 > \sigma_\theta \geq 17.5$
Slightly unstable	C	$17.5 > \sigma_\theta \geq 12.5$
Neutral	D	$12.5 > \sigma_\theta \geq 7.5$
Slightly stable	E	$7.5 > \sigma_\theta \geq 3.8$
Moderately stable	F	$3.8 > \sigma_\theta \geq 2.1$
Extremely stable	G	$2.1 > \sigma_\theta$

---

*Standard deviation of horizontal wind direction fluctuation over a period of 15 minutes to 1 hour.

APPENDIX A

FORMAT FOR HOURLY METEOROLOGICAL DATA TO BE PLACED ON MAGNETIC TAPE

Use: 9 track tape (7 will be acceptable)

Standard Label, which would include

Record Length = 160 characters

Block Size = 3200 characters (fixed block size)

Density = 1600 BPI preferred (800 BPI will be accepted)

Do not Use: Magnetic tapes with unformatted or spanned records.

(For guidance on tape formatting and a description of tape attributes, see Tables A-1 and A-2.)

At the beginning of each tape, use the first five records (which is the equivalent of ten cards) to give a tape description. Include plant name and location (latitude, longitude), dates of data, information explaining data contained in the "other" fields if they are used, height of measurements, and any additional information pertinent to identification of the tape. Make sure all five records are included, even if some are blank. Format for the first five records will be 160A1. Meteorological data format is (I6, I2, I3, I4, 25F5.1, F5.2, 3F5.1). Decimal points should not be included when copying data onto the tape.

All data should be given to a tenth of a unit except solar radiation, which should be given to a hundredth of a unit. This does not necessarily indicate the accuracy of the data (e.g., wind direction is usually given to the nearest degree, but record it with a zero in the tenth's place; therefore, 275 degrees would be 275.0 degrees and placed on the tape as 2750.) All nines in any field should indicate a lost record (99999). All sevens in a wind direction field should indicate calm (77777). If only two levels of data are monitored, use the upper and lower level fields. If only one level of data is monitored, use the upper level field.

TABLE A-1

MAGNETIC TAPE METEOROLOGICAL DATA

LOCATION:

DATE OF DATA RECORD:

I6	Identifier (can be anything)	
I2	Year	
I3	Julian Day	
I4	Hour (on 24-hr clock)	
		ACCURACY
F5.1	Upper Measurements: Level = ____ meters	
F5.1	Wind Direction (degrees)	____
F5.1	Wind Speed (m/s)	____
F5.1	Sigma Theta (degrees)	____
F5.1	Ambient Temperature (°C)	____
F5.1	Moisture: ____	____
F5.1	Other: ____	____
F5.1	Intermediate Measurements: Level = ____ meters	
F5.1	Wind Direction (degrees)	____
F5.1	Wind Speed (m/s)	____
F5.1	Sigma Theta (degrees)	____
F5.1	Ambient Temperature (°C)	____
F5.1	Moisture: ____	____
F5.1	Other: ____	____
F5.1	Lower Measurements: Level = ____ meters	
F5.1	Wind Direction (degrees)	____
F5.1	Wind Speed (m/s)	____
F5.1	Sigma Theta (degrees)	____
F5.1	Ambient Temperature (°C)	____
F5.1	Moisture: ____	____
F5.1	Other: ____	____

TABLE A-1 (Continued)

F5.1	Temp Diff (Upper-Lower) (°C/100 meters)	
F5.1	Temp Diff (Upper-Intermediate) (°C/100 meters)	
F5.1	Temp Diff (Intermediate-Lower) (°C/100 meters)	
F5.1	Precipitation (mm)	
F5.2	Solar Radiation (cal/cm^2/min)	
F5.1	Visibility (km)	
F5.1	Other:	
F5.1	Other:	

TABLE A-2

DATA TAPE ATTRIBUTES

USE		KEYWORDS	
		IBM	CDC
Mode:	9-track, 1600bpi, EBCDIC[1]	UNIT = TAPE9, DEN = 3	NT, PE, EB, S, CM = Yes
Internal Labels:	none[2]	LABEL = (;NL)	
Record Format:	fixed length/blocked	RECFM = FB	RT = F, BT = K
Record Length:	160 characters	LRECL = 160	FL = 160
Blocking:	3200 characters/block	BLKSIZE = 3200	RB = 20

DO NOT USE

Variable length or unformatted records or records that span tape blocks, e.g.:

IBM's RECFR = U or VBS.
CDC SCOPE standard tape data format (use the S parameter on the REQUEST to avoid this).

OTHER SYSTEMS

For systems other than IBM or CDC, the above information should be used as a guideline to produce tapes with similar characteristics.

---

[1] 9-track, 800 bpi, EBCDIC or 7-track, 800 or 556 bpi, BCD are also acceptable.

[2] IBM standard labels are also acceptable.

APPENDIX B

FORMAT FOR DATA INTERROGATION FROM METEOROLOGICAL SYSTEMS

To facilitate the remote interrogation requirements and the ability of the NRC to correctly access and use meteorological data, the procedures outlined below should be followed. A series of data bases should be coupled to executable codes to yield a file containing pertinent site information and selected meteorological data for emergency planning and for use during emergency situations. The access codes and execution instructions unique to the operational system should be documented and provided to the NRC and other appropriate organizations prior to implementation.

The information to be accessed should be available by executing an online program. Upon execution of the code, a query should be initiated requesting a meteorological data base starting and stopping time. The user response will be in the form of three free field entries composed of two time entries and the number of previous hours for which diffusion estimates are to be provided; i.e., YYJJJHHMM (starting time), YYJJJHHMM (stopping time), I (0 to 12), e.g., 801830015 801831200 12. A zero (0) or an all nine (999999999) field for stopping time should conclude the data set with the most recent set of observations and should continuously update for the duration of the log on session. The system response should include site descriptor, meteorological data field descriptor, and meteorological data from the previous I number of hours.

The information presented in the output file should identify site information contained within a 10-record block (mandatory filling of 10 lines with blanks, if necessary). These records should include the following items as a minimum: utility, plant names, plant location, elevation at the base of the meteorological tower, measurement heights above grade for meteorological parameters to be presented, and any additional information pertinent to identification of the site, tower, or parameters, e.g., last calibration date. The information should state whether the primary system is fully operational; if not, an indication should be given for those parameters that represent values recorded by the backup system. Suspect or lost data should be identified by the appropriate error code. The format for the site descriptor is given in Table B-1.

The meteorological information should be preceded by a 3-record block that provides a descriptor for each field of data. This 3-record block should be repeated for every 6-hour block of meteorological data, i.e., every 24 records. The format for the meteorological data field descriptor is given in Table B-2.

The meteorological data considered critical in emergency situations for initial estimation purposes should be provided by transmission. The list of parameters to be transmitted, which could be altered as procedures for evaluating the consequences of radioactive release change, should include 15-minute-averaged wind speed and direction at all measured levels, standard deviation of the horizontal wind direction fluctuations (σ_θ) at all measured levels, vertical temperature difference for all measured layers, ambient and dew point temperature at the 10-meter level, and the precipitation total for the 15-minute period.

All nines in any field should indicate a lost record or a parameter not monitored. All eights in any field should indicate the sensor is in place and recording, but, the information is deemed suspect. All sevens in the wind direction field should indicate calm. If only two levels of data are monitored, the upper and lower level fields should be used. If only one level of data is monitored, use the upper level field. The format for presentation of the meteorological data record is given in Table B-3.

TABLE B-1

SITE DESCRIPTOR DATA FORMAT
(8 Mandatory Records)

Record	Format	Content
1	80A1	Organization/Utility Name
2	80A1	Plant Name/Tower Identification
3	80A1	See Coding Form (Figure B-1)
4	F10.5	Latitude of Containment (degrees North)
	F10.5	Longitude of Containment (degrees West)
	F10.0	Elevation of Base of Meteorological Tower (feet above MSL)
	5X	Blank
	A10	Current Date
	A10	Current Time (session log on)
5	80A1	See Coding Form (Figure B-1)
6	F5.1	Height of Wind Sensor Upper Level (meters)
	F5.1	Height of Wind Sensor Intermediate Level (meters)
	F5.1	Height of Wind Sensor Lower Level (meters)
	5X	Blank
	F5.1	Upper Level Height of Temperature Difference (upper to lower measurement) (meters)
	F5.1	Lower Level Height of Temperature Difference (upper to lower measurement) (meters)
	5X	Blank
	F5.1	Upper Level Height of Temperature Difference (upper to intermediate measurement) (meters)
	F5.1	Intermediate Level Height of Temperature Difference (upper to intermediate measurement) (meters)
	5X	Blank
	F5.1	Intermediate Level Height of Temperature Difference (intermediate to lower measurement) (meters)
	F5.1	Lower Level Height of Temperature Difference (intermediate to lower measurement) (meters)
	5X	Blank
	F5.1	Height of Ambient Temperature Lower Level (meters)
	F5.1	Height of Dew Point Temperature Lower Level (meters)
	F5.1	Precipitation Gauge Height (meters)
7-10	80A1	Comment Section

TABLE B-2

METEOROLOGICAL DATA FIELD DESCRIPTOR
(3 Records for Every 6 Hours of Data)

Record	Format	Content
1	80X	Blank
2	80A1	See Coding Form (Figure B-2)
3	80X	Blank

TABLE B-3

METEOROLOGICAL DATA FORMAT
(1 Record Per 15-Minute-Averaged Data Set)

Format	Content
I2	YEAR
I3	JULIAN DATE
I2	HOUR (on 24-hour clock)
I2	MINUTE (ending observation)
F4.0	WIND DIRECTION (degrees)* UPPER LEVEL
F4.0	WIND DIRECTION (degrees)* INTERMEDIATE LEVEL
F4.0	WIND DIRECTION (degrees)* LOWER LEVEL
1X	BLANK COLUMN
F4.1	WIND SPEED (m/s) UPPER LEVEL
F4.1	WIND SPEED (m/s) INTERMEDIATE LEVEL
F4.1	WIND SPEED (m/s) LOWER LEVEL
1X	BLANK COLUMN
F3.0	SIGMA THETA (degrees) UPPER LEVEL
F3.0	SIGMA THETA (degrees) INTERMEDIATE LEVEL
F3.0	SIGMA THETA (degrees) LOWER LEVEL
1X	BLANK COLUMN
F5.1	TEMPERATURE DIFFERENCE (°C/100 m) UPPER-LOWER
F5.1	TEMPERATURE DIFFERENCE (°C/100 m) UPPER-INTERMEDIATE
F5.1	TEMPERATURE DIFFERENCE (°C/100 m) INTERMEDIATE-LOWER
1X	BLANK COLUMN
F5.1	AMBIENT TEMPERATURE (°C) LOWER LEVEL
1X	BLANK COLUMN
F5.1	DEW POINT TEMPERATURE (°C) LOWER LEVEL
1X	BLANK COLUMN
F5.1	PRECIPITATION TOTAL (mm) GROUND LEVEL
1X	BLANK COLUMN
I1	PASQUILL STABILITY CLASS OR EQUIVALENT TO BE ASSUMED FOR DIFFUSION ESTIMATES (1 = A, 2 = B, 3 = C, ..., 7 = G)

---

*Wind direction indicates the direction _from_ which the wind is coming.

NRC FORM 53A
(12-75)

U. S. NUCLEAR REGULATORY COMMISSION
ADP TRANSCRIPTION SHEET

JOB TITLE ______________________

OFFICE ______________________

PAGE ______ OF ______

DATE ______________________

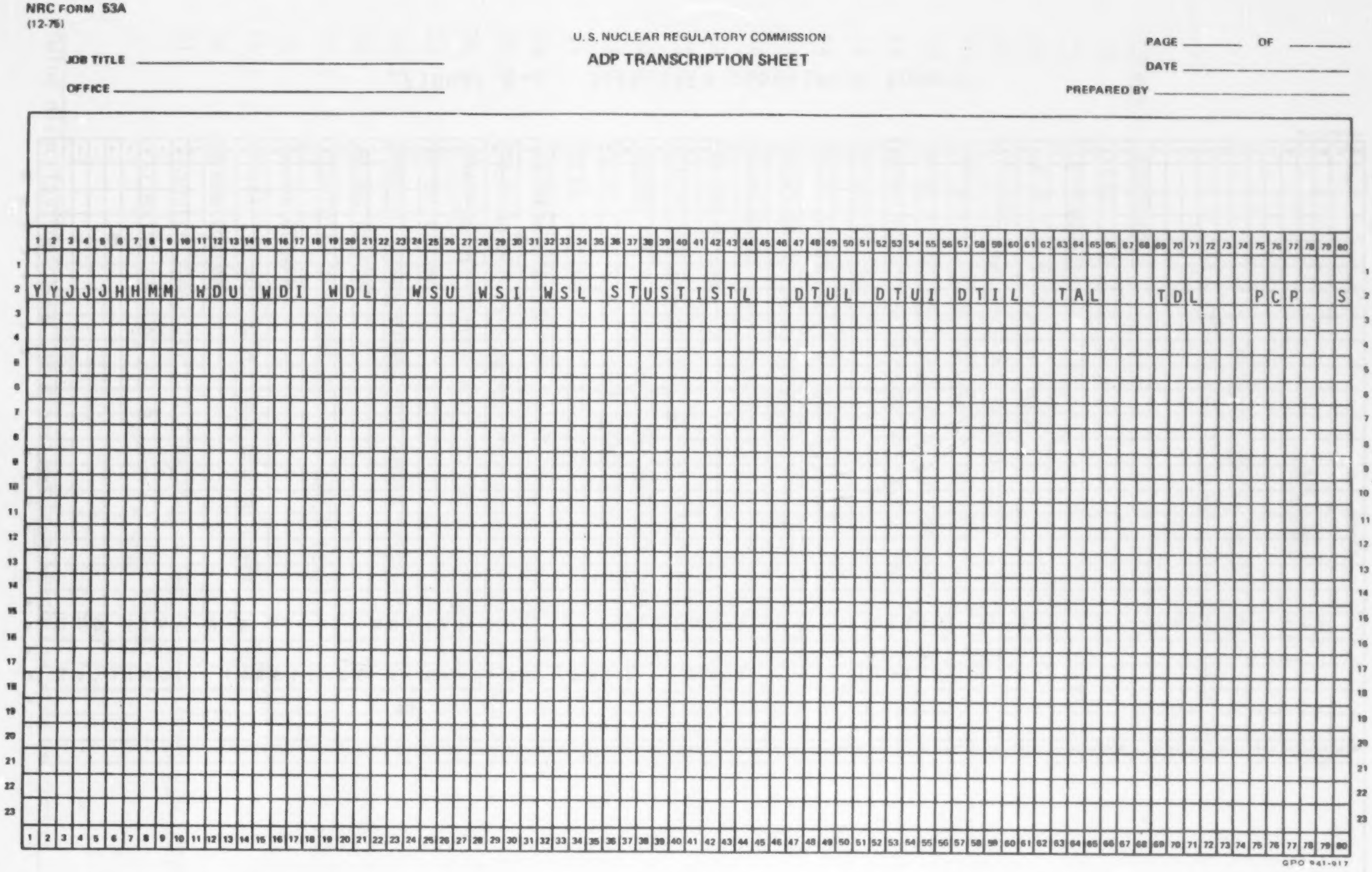
PREPARED BY ______________________

```
 LATITUDE  LONGITUDE ELEVATION (MSL)   DATE      TIME

WIND SENS U I L     DELTAT U-L     DELTAT U-I     DELTAT I-L       TA  TD  PCP
```

GPO 941-917

FIGURE B-1. SITE DATA DESCRIPTOR FORMAT



NRC FORM 53A
(12-76)

U. S. NUCLEAR REGULATORY COMMISSION

ADP TRANSCRIPTION SHEET

JOB TITLE ______________________

OFFICE ______________________

PAGE ________ OF ________

DATE ______________________

PREPARED BY ______________________

```
YYJJJHHMM WDU WDI WDL  WSU WSI WSL STUSTISTL  DTUL DTUI DTIL  TAL   TDL   PCP  S
```

GPO 941-917

FIGURE B-2. METEOROLOGICAL DATA FIELD DESCRIPTOR FORMAT

DRAFT VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

The proposed action consists of the revision of Regulatory Guide 1.23 (Safety Guide 23) to update the recommendations for establishing and operating the meteorological programs at nuclear power plant sites. The meteorological programs are necessary to measure and collect meteorological information that is used in estimating potential radiation doses to the public resulting from actual routine releases of radioactive materials into the atmosphere and to estimate either potential doses to the public as a result of a hypothetical reactor accident or actual doses in the case of a real accident.

1.2 Need for Proposed Action

Regulatory Guide 1.23, "Onsite Meteorological Programs," was originally issued as Safety Guide 23 in February 1972. This guide has never been revised. Consequently, much of the information provided in the guide is obsolete, having been made so by changes in the state of the art in meteorological measurement technology and by changes discussed in the guide in the meteorological evaluation procedures in which the meteorological data are to be used. In addition, recent staff experience during the accident at Three Mile Island has shown that a capability to access the meteorological data remotely should be an integral part of the operational meteorological measurement program. The present version of the guide does not include such a recommendation. A revision of this guide is deemed necessary, therefore, to strengthen the guidance in an area shown to be weak as a result of the Three Mile Island experience and to update other areas that are obsolete and of little or no value to users.

1.3 Value/Impact of Proposed Action

1.3.1 NRC

Completion of the proposed action is estimated to require from 0.5 to 1.0 man-year of effort. Associated costs include printing and copying costs and costs for normal office supplies. No additional research or technical assistance contract costs in support of this effort are anticipated. However, the possibility exists that unanticipated future developments may indicate the advisability of efforts that would incur additional manpower, contract, and travel costs. The primary benefit to the NRC would be a reduction in the number of inquiries that arise because of the obsolete information in the unrevised guide, which does not adequately describe the present recommendations for meteorological programs. Finally, if the recommendation in the proposed revision concerning the remote-access capability for meteorological data is implemented, the NRC will have the potential to acquire real-time meteorological data if needed in the event of an emergency.

1.3.2 Other Government Agencies

Applicant agencies (e.g., TVA) would be affected as discussed in Section 1.3.3. Additional workload would accrue at other agencies (e.g., NOAA, EPA) that may participate in terms of review and comment services. Upon completion of the proposed action, other agencies will have a current reference document describing the NRC's recommendations concerning meteorological measurement programs at nuclear power plant sites. Should the remote-access capability recommendation be implemented, agencies (Federal, State, and local) that may be involved in an emergency situation because of an accident at a nuclear power plant will be assured of access to meteorological data representative of the site in the event that data-recording instruments become inaccessible by conventional means.

1.3.3 Industry

Industry will benefit by having available a current source of information concerning NRC recommendations for establishing and operating an onsite meteorological program at nuclear power plant sites. Costs to industry will result from having to become familiar with the product document and in review and comment efforts. It is anticipated that studies will cost approximately \$100,000 to \$200,000 at those coastal sites where it is necessary to experimentally confirm the heights of the internal boundary layer. The total estimated cost of purchasing and installing the necessary equipment for a remote-access capability to archive meteorological data for emergency response purposes, should this recommendation be implemented, is \$10,000 to \$20,000 per site. This equipment is expected to be part of a larger system for making atmospheric dispersion calculations and dose assessments. Annual operating costs are estimated to be less than \$2,000. In some cases, the cost may be offset because this equipment can also be used as part of the meteorological data reduction system. The benefit to industry resulting from the installation of such a capability will be the ability to provide emergency response groups involved in an accident situation with meteorological data representative of the site, even if the onsite recorders are inaccessible.

1.3.4 Public

The public will bear the monetary costs of completing and implementing the proposed action. In addition, the costs incurred by the utilities from the installation and operation of the remote-access capability, if implemented, would be expected to be passed on to the consumers of electric power in the form of higher rates. In return, the public will benefit by an increased assurance that meteorological information representative of the site, which might prove crucial in an emergency situation, will be available to emergency response groups. The public will also benefit from the availability of a current reference document that presents the NRC's recommendations concerning meteorological measurement programs at nuclear power plant sites.

1.4 Decision on Proposed Action

The proposed action should be accomplished on a priority basis.

2. TECHNICAL APPROACH

The alternative methods of accomplishing the proposed action are to perform the work in-house or initiate a technical assistance contract with an independent contractor.

2.1 Discussion and Comparison of Technical Alternatives

The information and expertise needed to revise the guide is currently available within the NRC. The amount of work necessary to accomplish the proposed action is of limited extent and can be performed in-house within the anticipated time frame without adversely impacting on other task requirements. Considerable time would be expended on the initiation and completion of a technical assistance contract with an independent contractor. Although staff time expended on direct work on the proposed action would be eliminated by contracting the task, additional staff time would be required to prepare and issue a contract and monitor contractor performance.

2.2 Decision on Technical Alternatives

Since the information and expertise to accomplish the proposed action exists within the NRC, the completion of the task in-house is the most beneficial technical alternative.

3. PROCEDURAL APPROACH

Since the proposed action is an update of information contained in an existing regulatory guide, the only appropriate procedural approach is a revision to the existing guide.

4. STATUTORY CONSIDERATIONS

4.1 NRC Authority

Authority for this guide would be derived from the safety requirements of the Atomic Energy Act through the Commission's regulations. In particular, paragraph 100.10(c)(2) of 10 CFR Part 100 states that, in determining the acceptability of a site for a power or test reactor, the Commission will take into consideration meteorological conditions at the site and in the surrounding area. Appendix E, "Emergency Plans for Production and Utilization Facilities," to 10 CFR Part 50 requires that applicants for an operating license develop plans for coping with radiological emergencies. The plans must include criteria for determining when protective measures should be considered within and outside the site boundary to protect health, safety, and property. In this regard, it is necessary for the applicant to establish and maintain a meteorological program capable of rapidly assessing critical meteorological parameters. Further, paragraph 50.36a(a)(2) of 10 CFR Part 50 requires nuclear power plant licensees to submit semiannual reports specifying the quantity of each of the principal radionuclides released to unrestricted areas in gaseous effluents and such other information as may be required by the Commission to estimate maximum potential doses to the public resulting from these releases to ensure compliance with the requirements of 10 CFR Part 20. A knowledge of meteorological conditions in the vicinity of the plant is necessary to make these estimates. Finally, in order for the Commission to fulfill its responsibilities under NEPA and in accordance with the requirements of Appendix I to 10 CFR Part 50 and of 10 CFR Part 51, meteorological information must be available for use in assessing potentially adverse environmental effects resulting from the construction or operation of a nuclear power plant.

4.2 Need for NEPA Assessment

The proposed action is not a major action as defined by paragraph 51.5(a)(10) of 10 CFR Part 51 and does not require an environmental impact statement.

5. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICIES

No potential conflicts with other agencies have been identified. The product document will be used in the implementation of 10 CFR Part 20, 10 CFR Part 50, 10 CFR Part 51, and 10 CFR Part 100 as described above. The guidance in the proposed revision will be consistent with that in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations," Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled

Reactors," and Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants."

Regulatory Guide 3.8, "Preparation of Environmental Reports for Uranium Mills," references the meteorological measurement program and data format presented in Regulatory Guide 1.23. Since the revised meteorological measurement programs described in proposed Revision 1 to Regulatory Guide 1.23 may not be appropriate for most uranium mills, a further revision of Regulatory Guide 3.8 may be needed. Additional expenditure of manpower to make changes is expected to be small because Regulatory Guide 3.8 is currently in the final process of comment resolution.

Backfitting requirements will result from the implementation of the remote-access capability at those installations where the data reduction systems do not presently have such a capability. Potential backfitting costs are discussed above.

6. SUMMARY AND CONCLUSIONS

A revision to Regulatory Guide 1.23, "Onsite Meteorological Programs," should be prepared. This revision should be done in-house.

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OFFICE OF STANDARDS DEVELOPMENT
DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

September 1980
Division 1
Task RS 807-5

Contact: S. D. Richardson (301) 443-5913

SECOND PROPOSED REVISION 2[*] TO REGULATORY GUIDE 1.8
PERSONNEL QUALIFICATION AND TRAINING

A. INTRODUCTION

Paragraph 50.34(b)(6)(i) of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), requires that applications for a license to operate a nuclear power plant include information concerning organizational structure, personnel qualifications, and related matters. Paragraph 55.10 of 10 CFR Part 55, "Operators' Licenses" (Ref. 2), requires that operator license applications include information concerning an individual's education, experience, and related matters. This regulatory guide describes a method acceptable to the NRC staff for complying with these portions of the Commission's regulations with regard to the qualification and training of nuclear power plant personnel filling the various functional positions described in the December 6, 1979, draft version of American Nuclear Society (ANS) Standard ANS 3.1, "Qualification and Training of Personnel for Nuclear Power Plants"[1] (Ref. 3).

FOR COMMENT

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[*] The substantial number of changes from the first proposed Revision 2 to this guide (Task RS 807-5) dated February 1979 have made it impractical to indicate the changes in the margin. Since the issuance of the first proposed Revision 2 to Regulatory Guide 1.8 for public comment, much guidance concerning personnel qualification and training has been developed through assessment of the Three Mile Island Unit 2 accident by various organizations. In addition, ANSI/ANS 3.1-1978, which is endorsed by the regulatory guide, is undergoing extensive revision in an effort to provide upgraded requirements for personnel qualifications and training. As a result of the incorporation of additional guidance into the revisions of ANSI/ANS 3.1 and the regulatory guide, this second proposed Revision 2 to Regulatory Guide 1.8 endorsing the December 6, 1979, draft version of ANS 3.1 is being issued for public comment in order to obtain additional public input on the proposed regulatory guidance.

[1] Copies of Draft Standard ANS 3.1 and preceding versions of it, as well as other ANS standards cited in this guide, are available from the American Nuclear Society, 555 North Kensington Avenue, LaGrange Park, Illinois 60525.

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by DEC 5 1980

Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

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B. DISCUSSION

1. DEVELOPMENT OF GUIDE

1.1 Impact of Events

In the aftermath of the Three Mile Island nuclear plant accident (TMI-2), the industry, the NRC, and others have conducted a number of studies and investigations, and, as a result of their findings, have recommended changes in the numbers of nuclear power plant personnel now employed, and their qualifications and organization. The principal studies have been conducted by the President's Commission on Three Mile Island (Ref. 4), the NRC Special Inquiry Group (Ref. 5), the Office of Inspection and Enforcement Special Review Group (Ref. 6), the Office of Nuclear Reactor Regulation's (NRR) Lessons Learned Task Force (Refs. 7 and 8), and NRR's Bulletins and Orders Task Force (Ref. 9).[2] Collectively, these studies have called for a general upgrading of utility capabilities for handling routine plant operations and for coping with unusual or unexpected conditions. As might be expected, the recommendations of these diverse groups are not completely compatible; what is clear, however, is that all of these groups have called for upgrading in at least two identifiable areas: management oversight and technical competence. Optimum qualification and training of nuclear power plant personnel is necessary to upgrade these areas.

1.2 Relationship to National Standards Effort

Revision continues to existing standards for qualification and training of nuclear power plant personnel. Based on existing knowledge and experience in 1971, Subcommittee ANS-3, Reactor Operations, of the American Nuclear Society Standards Committee developed a standard containing criteria for the qualification and training of nuclear power plant personnel. This standard was approved by the American National Standards Institute (ANSI) Committee N18, Design Criteria for Nuclear Power Plants, and designated ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel" (Ref. 10). Regulatory Guide 1.8, "Personnel Selection and Training," endorsing ANSI Standard N18.1-1971 was issued in March 1971. A revision to this standard was subsequently approved by the ANSI Board of Standards Review and designated ANSI/ANS 3.1-1978, "Selection and Training of Nuclear Power Plant Personnel" (Ref. 11).

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[2] The recommendations of the investigating groups are collected in NUREG-0660, "NRC Action Plan Developed As A Result of the TMI-2 Accident." NUREG-0660, in Appendix E, discusses the availability of the individual investigatory reports (also see References).

In February 1979, a draft Revision 2 of Regulatory Guide 1.8 (RS 807-5) endorsing ANSI/ANS 3.1-1978 was issued for public comment. In May 1979, additional comments on the area of personnel qualifications were requested in light of experience gained from the TMI-2 accident. Copies of all comments received both on the February 1979 draft guide and as a result of the May 1979 request for comments were provided to the ANS-3 Subcommittee for their use during development of a revision to the 1978 standard. The draft revision incorporating upgraded requirements was approved by the ANS-3 Subcommittee on December 6, 1979. Draft Standard ANS 3.1 has been extensively revised in most areas of the standard on which comments were received. Public comments are now being solicited on this second proposed Revision 2 to Regulatory Guide 1.8 that endorses the revised standard.

The significant changes incorporated into the revised standard include (a) a reformulation of Section 4, "Qualification," to separate the topics of education, experience, and training in order to more clearly define personnel qualification requirements; (b) an upgrading of a number of specific qualification requirements, particularly in the area of education and experience requirements; (c) more definitive guidance on training and retraining programs including requirements for the use of position task analyses to define required training and for the use of simulators in training and retraining programs; (d) a listing of additional specific control manipulations to be performed during the retraining program; (e) requirements for corporate management certification of individuals prior to licensing examination by the NRC, and (f) qualification requirements for individuals directing preoperational and startup tests.

2. ONGOING EFFORTS AFFECTING GUIDE

2.1 NRC Philosophy and Objectives

NUREG-0660, "NRC Action Plan Developed As A Result of the TMI-2 Accident" (Ref. 12),[2] in Chapter I, "Operational Safety," describes actions intended to substantially improve and emphasize operational safety, an area that has not previously been given the same regulatory emphasis as nuclear power plant design. The actions have two complementary objectives: (1) to reduce challenges to the safety of the plant and (2) to ensure proper reactions to challenges that do occur. Both of these objectives can only be achieved by requiring optimum qualification and training of nuclear power plant personnel. The reduction of challenges requires a highly qualified staff that devotes unflagging attention to the proper operation of the plant, continuous monitoring to verify that plant operations are correctly performed, and correcting and improving operations by the feedback of operating experience to appropriate personnel. The proper reaction to challenges to the safety of the plant requires a thorough understanding of plant design and plant response to unusual conditions, as well as training in the diagnosis of these conditions and reaction to them.

Changes needed in this regulatory guide are discussed in two items of the NRC Action Plan (NUREG-0660): Item I.A.2.6, "Long-Term Upgrading of Training and Qualifications," and Item I.B.1.1, "Organization and Management of Long-Term Improvements." Short-term actions related to personnel qualifications already taken by the NRC, as well as changes included in the December 6, 1979, draft version of ANS 3.1, "Qualification and Training of Nuclear Power Plant Personnel," are incorporated in this second proposed Revision 2 to Regulatory Guide 1.8.

2.2 Description of Efforts

As indicated in NUREG-0660, there are a number of efforts currently in progress to provide additional reviews and recommendations concerning the subject of personnel qualification and training that interface with this regulatory guide. The principal efforts in progress within the NRC on personnel qualification and training are described in Sections 2.2.1 through 2.2.8 below.

2.2.1 Upgrading Qualifications of Operators

The NRC staff provided recommendations in Commission Paper SECY 79-330E, "Qualifications of Reactor Operators" (Ref. 13), for upgrading the qualifications of licensed operators and senior operators through a program that includes increased training and testing in the areas of thermal-hydraulics and reactor transient response, increased use of simulator training and testing, higher passing grades on licensing examinations, increased experience requirements for operators and senior operators, and increased emphasis on retraining and examination. Based on Commission action on SECY 79-330E, the Office of Nuclear Reactor Regulation by a letter of March 28, 1980, from H. R. Denton to all power reactor applicants and licensees (Ref. 14) set forth the revised criteria to be used by the staff in evaluating reactor operator training and licensing and established effective dates for their implementation. Licensees and applicants were also informed in the March 28, 1980, letter of other criteria under development that would require additional staff work to produce the necessary requirements that would eventually be established through rulemaking proceedings. Additionally, licensees and applicants were informed that Commission review in the area of operator training and qualification is continuing and may result in identification of additional criteria to be used in evaluation of operators for licensing and relicensing.

2.2.2 Recommended Revisions to Part 55 and Part 50

The NRC staff is preparing amendments to 10 CFR Part 55, "Operators' Licenses," and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," that incorporate applicable Commission-approved recommendations of SECY 79-330E and other proposed changes resulting from additional staff review of Part 55 and Part 50. The amendments focus on education and training for prospective operators, requirements for simulator training for operators and operator applicants, NRC participation in requalification examinations, operator understanding of the theory behind operation of a facility, limiting extension of license expiration dates, and maintenance of operator proficiency at operating the facility controls. The Regulatory Activities Subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) has reviewed the proposed changes to Parts 55 and 50.[3] Both the March 28, 1980, H. R. Denton NRR letter and the proposed rulemaking impact this proposed regulatory guide. With respect to the March 28, 1980, letter, the short-term requirements are either incorporated in the December 6, 1979, Draft Standard ANS 3.1 or noted in the regulatory position of this regulatory guide. With respect to Part 55 and Part 50 revisions, certain positions now included in this regulatory guide will be codified in the regulations in place of the existing general provisions of the regulations. After publication of the final rule changes, Regulatory Guide 1.8 will be revised to eliminate any duplication.

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[3] A transcript of their review dated June 4, 1980, is available for inspection or copying for a fee in the NRC Public Document Room at 1717 H Street NW., Washington, D.C.

One aspect of changes recommended by SECY 79-330E and applicable only to NRC that is not addressed in proposed changes to 10 CFR Part 55 or this regulatory guide concerns NRC practices for administration of qualifying examinations for nuclear power plant personnel. These practices include the following items:

a. Time limits will be imposed for completion of the written operator and senior reactor operator examinations administered by the NRC. The operator examination will be limited to 9 hours and the senior operator portion of the examination will be limited to 7 hours.

b. The passing grade for the operator and senior operator examination will be 80% overall and 70% in each category.

c. The NRC staff will audit training programs more closely and administer all the certification exams conducted at the completion of the simulator training portion of an operator's training program.

d. The NRC staff will administer or direct the facility to administer annual examinations as part of the licensed operator and senior operator requalification programs that include a written, oral, and practical (performed on a simulator) portion.

e. As a condition of license application, applicants for operator and senior operator licenses will be required to grant permission to the NRC to inform their facility management of the results of their examinations.

2.2.3 Onsite and Offsite Managerial and Technical Organizations

The NRC staff has under development criteria for onsite and offsite managerial and technical organizations, including a radiological protection organization, that will provide assurance of the safe operation of the plant during normal and abnormal conditions and the capability necessary to respond to unusual or unexpected situations. A contractor, Teknekron Research, Inc., was selected to assist in the development of criteria listed above. Teknekron submitted its final report to the NRC in May 1980 (Ref. 15).

Task I.B.1.1, "Organization and Management of Long-Term Improvements," and Task I.B.1.2., "Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants," of NUREG-0660 describe the criteria development effort for technical and managerial organizations. Task I.B.1.2 is a first step in the development of criteria described in Task I.B.1.1. As described in Task I.B.1.2, near-term operating licensee applicants, as part of the license application review process, are being required to comply with the findings and requirements generated in an NRC interoffice review of licensee organization and management. The review is based, in part, on an NRC document entitled "Draft Criteria for Utility Management and Technical Competence" (Ref. 16). The first draft of this document was dated February 25, 1980, however, the document continues to change with use and experience in ongoing reviews. The document addresses the organization, resources, training, and qualifications of plant staff and management (both onsite and offsite) for routine operations and accident conditions. The criteria developed for use in evaluation of operator license applicants, as modified by experience from interoffice review of near-term operating license applicants, will form the basis for the final criteria to be developed in accordance with Task I.B.1.1 of NUREG-0660. The criteria reference Regulatory Guide 1.8 and ANS Standard 3.1 with regard to qualifications of nuclear plant personnel filling the various functional positions described in ANS Standard 3.1. Regulatory Guide 1.8 will be implemented in conjunction with the recommendations of Task I.B.1.1 of the NRC Action Plan (see Section D of this guide, Implementation).

2.2.4 Onsite "Independent Safety Engineering Group"

Each near-term operating license applicant as a part of ongoing license application review is being required to establish a group, commonly referred to as the "Independent Safety Engineering Group," that is independent of the plant staff, but is assigned onsite to perform independent reviews of plant operations activities and to evaluate operating experiences at nuclear power plants. Commission Information Paper SECY-80-242 (Ref. 17) describes the relationship of the proposed "Independent Safety Engineering Group" to other organizational entities such as the Plant Operations Review Committee. The "Draft Criteria for Utility Management and Technical Competence" (see Item 2.2.3 above) addresses the qualifications of the onsite independent safety engineering group by referencing Section 4.7 of the December 6, 1979, Draft Standard ANS 3.1.

2.2.5 Shift Technical Advisors

As described in Task I.A.1.1 of NUREG-0660, the NRC is requiring that a technical advisor to the shift supervisor be present on all shifts.[4] Although minimum training requirements have not yet been specified for shift technical advisors presently occupying such positions, shift technical advisors should enhance the accident-assessment function at the plant. By January 1, 1981, shift technical advisors will be required to have a technical education, which is of the quality of courses taught at the college level and includes about 60 semester hours in basic subjects of engineering and science; specific training in the design, function, arrangement, and operation of plant systems; and training in the expected response of the plant and instruments to normal operation, transients, and accidents, including multiple failures of equipment and operator errors. Commission Information Paper SECY 80-243 (Ref. 18) addresses the subject of shift technical advisors, including a summary of experience required, duties to be performed, and the number of shift technical advisors required onsite. The qualifications of shift technical advisors as addressed in this regulatory guide are the same as those specified in Commission Paper SECY 80-243. The Commission was informed in this paper that the Institute of Nuclear Power Operations (INPO) has recently forwarded to NRR for comment a copy of an INPO document, "Nuclear Power Plant Shift Technical Advisor--Recommendations for Position Description, Qualifications, Education and Training" (Enclosure to SECY 80-243), and that the NRC staff had the document under review. This review may lead to revision to this regulatory guide with regard to shift technical advisors.

2.2.6 Comparison of NRC, Commercial, and Naval Procedures for Qualification of Personnel

In early 1980, Basic Energy Technology Associates, Inc. (BETA), completed a study for the Office of Nuclear Reactor Regulation that outlined the results of a comparative review of current NRC requirements, commercial nuclear power plant practices, and the Naval Nuclear Propulsion Program procedures for the selection, training, and qualification of personnel involved in nuclear power plant operation and maintenance. The results of the BETA study entitled "Power Plant Staffing" are documented in NUREG/CR-1280, BETA-103 (Ref. 19). Public comments were requested on this study. The BETA study, when comparing the majority of practices, used the provisions outlined in ANSI N18.1-1971 and did not consider later versions of the standard. The BETA study will be considered when developing future revisions to this regulatory guide.

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[4] Requirements for the Shift Technical Advisor were contained in H. R. Denton and D. G. Eisenhut NRR letters to operating plant licensees on September 13 and October 30, 1979; applicants for pending operating licenses on September 27 and November 9, 1979; and applicants for pending construction permits and licensees of plants under construction on October 10 and November 9, 1979. Copies of these letters are available for inspection or copying for a fee in the NRC Public Document Room at 1717 H Street NW., Washington, D.C.

2.2.7 Requirements for Licensing of Operators

The NRC staff has awarded a contract to Analysis and Technology, Inc., for a study of requirements for operator licensing.[5] The scope of the work should result in recommendations for (a) the means to be employed for selection and training of nuclear power plant personnel and the degree to which the NRC should be involved in the process, (b) the means to be employed to evaluate the effectiveness of training programs, including who, by job description, should be licensed, (c) the methods to be employed to ensure continued competency of plant personnel and the degree to which the NRC should be involved in the methods, (d) the methods to be employed for maintaining a highly motivated and dedicated work force, and (e) the means to be employed for rapidly requalifying presently licensed operators to meet the proposed new requirements. Task I.A.2.6, "Long-Term Upgrading of Training and Qualifications," of NUREG-0660 provides a description of planned NRC actions related to this contract study. After staff review of the study has been completed, the staff will provide recommendations to the Commission and subsequently factor Commission decisions on the recommendations into a regulatory guide or a regulation.

2.2.8 Planned NRC Accreditation of Training Institutions

As described in Task I.A.2.7, "Accreditation of Training Institutions," of NUREG-0660, the NRC staff is conducting a study of procedures and requirements for NRC accreditation of training institutions. A Commission Information Paper on this subject should be completed by late 1980. The NRC staff will also prepare a Commission paper examining various NRC approaches to accreditation of training institutions. Staff action on this later effort should be completed by January 1982.

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[5] Since the study is in progress, its recommendations are not yet available.

3. ANTICIPATED REVISION TO GUIDE

With regard to the interface of this proposed regulatory guide and the other efforts described in Section 2.2 above, this version of the regulatory guide is a first step in a comprehensive process of upgrading the training and qualifications of operations personnel. Some of the regulatory positions may be eventually superseded by more comprehensive long-term changes (e.g., changes to NRC regulations) in the area of personnel qualification and training; however, these positions are expected to be consistent with the long-term changes.

This regulatory guide is being issued at this time in order to solicit public comment on changes thus far made. The draft standard, ANS 3.1, endorsed by this guide has not yet been approved as an American National Standard, but permission has been granted by the American Nuclear Society to use the December 6, 1979, draft revision of the standard with the proposed regulatory guide revision during the public review and comment period. Comments received on the draft standard will be transmitted to the ANS-3 Working Group that is developing the standard in addition to being given consideration by the staff for inclusion in the regulatory guide. Since the standard and the guide are being developed in parallel, comments can be resolved, as appropriate, by modification to either the standard or the guide or both.

C. REGULATORY POSITION

The requirements included in Draft Standard ANS 3.1 dated December 6, 1979, are acceptable to the NRC staff for complying with the pertinent requirements of the Commission's regulations for qualification and training of personnel in the operations organization subject to the following exceptions and supplements. (The discussions in Appendix A related to Regulatory Position 2.3.1 about Shift Supervisors and in Appendix B related to Regulatory Position 2.7 on Shift Technical Advisors should be considered with those positions.)

1. GENERAL: EXCEPTIONS AND SUPPLEMENTS

1.1 Pertinent Standards and Endorsing Guides

Throughout the draft standard, other documents required to be included as part of this standard are identified at the point of reference. The specific applicability of these standards listed in the draft standard has been addressed in the latest revision of the following regulatory guides:

Regulatory Guide	ANSI Standard[1]
1.17	N18.17
1.33	N18.7
1.134	N546

In addition, the NRC has issued for comment Draft Guide RS 110-5, "Nuclear Power Plant Simulators for Use in Operator Training," dated July 1980, endorsing the March 24, 1980, draft version of American Nuclear Society Standard ANS 3.5, "Nuclear Power Plant Simulators for Use in Operator Training."[1]

1.2 Temporary Personnel Replacements

1.2.1 Field-Specific Experience

Section 3.1 of Draft Standard ANS 3.1 discusses the use of personnel who may not meet the requirements of the standard to fill positions as temporary replacements. Temporary personnel should not be used as replacements for periods exceeding 1 month. Additionally, temporary personnel replacements should have experience in the field of the individual for whom they are serving as replacements.

1.2.2 Training

When an individual is hired to temporarily function as a plant employee, such as for contracted services, evidence of previous education, experience, and training should be provided and reviewed by the appropriate professional-technical group leaders. The appropriate group leaders should then determine the content for that individual's training, including plant-specific training. As a minimum, each individual should receive "General Employee Training" (see Section 5.4 of ANS 3.1).

1.3 Definition of "College-Level Education"

Throughout the standard, the term "college-level education" is used to describe the quality of the education to be completed; i.e., courses completed should be as demanding as those offered at the college level. This term should be construed to mean course work satisfactorily completed (e.g., finished with a grade at the 70% level) at or conducted by a college or university with curricula accredited by a nationally recognized agency such as the Accreditation Board for Engineering and Technology (ABET/ECPD).[6] Additional guidance on accreditation of training institutions[7] will be developed by the NRC. The NRC intends to establish alternative means of accreditation of training programs, after which, course work completed under such programs may be acceptable.

1.4 Interim Regulatory Positions Related to Anticipated Rules

Section 1 of Draft Standard ANS 3.1 states that the NRC and other regulatory agencies promulgate regulations applying to many aspects of the design, construction, and operation of nuclear power reactors and that ANS 3.1 shall not take precedence over any such regulation. As indicated in the Section B "Discussion" of this guide, the NRC staff is preparing amendments to 10 CFR Part 55 and 10 CFR Part 50 that relate to subjects addressed by ANS 3.1. Listed below are exceptions taken to the standard to provide for consistency with the anticipated amendments. The proposed amendments to 10 CFR Part 55 and 10 CFR Part 50 are undergoing review prior to publication for public comment. The positions listed below should be regarded as interim regulatory positions pending issuance of final amendments to 10 CFR Part 55 and 10 CFR Part 50.

a. Section 4.3.1 of Draft Standard ANS 3.1 requires as part of the experience requirements that a senior operator have 6 months of experience as a licensed operator. Except in cases where personnel are taking cold NRC senior operator examinations, a senior operator should have 1 year of experience as a licensed operator at the plant for which the senior operator license is requested.

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[6] The Manual of Evaluation Procedure of the Engineering Accreditation Commission may be obtained from the Accreditation Board for Engineering and Technology (ABET/ECPD), 345 East 47th Street, New York, New York 10017.

[7] The term "training institutions" includes colleges, universities, other named four-year schools of higher learning, and technical schools that offer courses of the quality and at the level of comprehension usually taught in colleges.

b. Sections 4.3.1.2 and 4.5.1.2 require that the competency of license applicants for senior operator and operator licenses be certified by corporate management prior to endorsing the applicant for licensing by the NRC. This certification should be performed by the highest level of corporate management responsible for plant operations (for example, the Vice-President for Operations). Additionally, documented evidence of this certification should be submitted with the license application and this documentation retained for at least as long as that individual is employed at the plant or by the organization that operates the plant.

c. Section 4.3.1.2, in discussing education requirements, states that a senior operator shall have a high school diploma plus the equivalence of thirty (30) semester hours of college-level education (450 classroom or instructor-conducted hours) in mathematics, reactor physics, chemistry, materials, reactor thermodynamics, fluid mechanics, heat transfer, electrical and reactor control theory. In lieu of this provision, a senior operator should hold a high school diploma or general education development certificate and have had a minimum of 60 semester hours of college-level education in technical subjects such as mathematics, reactor physics, chemistry, materials, reactor thermodynamics, fluid mechanics, heat transfer, and electrical and reactor control theory.

d. Section 5.2.1.3.1 requires that, as part of the operator training program, candidates shall observe operating practices in the control room.

As a minimum, an operator license applicant should have received 3 months of shift training with no other concurrent duties at the facility for which the applicant seeks a license. During this training, the applicant, under the observation and control of a licensed operator, should have manipulated the facility controls and performed duties that would be performed by a licensed operator.

As a minimum, a senior operator applicant should have received 3 months of shift training with no other concurrent duties at the facility for which the applicant seeks a license. During this training, the applicant, under the observation and control of a licensed senior operator, should have supervised the manipulation of the facility controls and performed duties that would be performed as a licensed senior operator.

e. Section 5.5.1, in discussing retraining of licensed personnel, states that a simulator shall be used to fulfill portions of the retraining program for those evolutions where the simulator is capable of simulating continuously and in real time plant operations of the referenced facility. Proposed amendments to Part 55 are expected to contain requirements for mandatory simulator training for both new license applicants and requalification of operators and senior operators. Licensees should provide, to the extent possible, for simulator training.

2. QUALIFICATIONS: EXCEPTIONS AND SUPPLEMENTS

2.1 Limited Number of Exceptions to Required Qualifications

Section 4.1 discusses a case-by-case evaluation of an individual's qualifications when the individual does not meet those stated in the standard. Acceptance of an individual's qualifications based on this type of evaluation should be reserved for exceptional cases. Exceptions for individual qualifications should not exceed 5 percent of all the positions covered by this standard on a per unit basis without prior NRC staff approval. Exceptions should not be used for the positions of plant manager, operations manager, radiation protection manager,[8] or shift supervisor. In those cases where exceptions to requirements of the standard are used, the evaluation that justifies any such exceptions should be documented and that documentation retained for at least as long as that individual is employed at the plant or by the organization that operates the plant.

2.2 Managers

2.2.1 Plant and Technical: NRC Certification

Sections 4.2.1 and 4.2.4 discuss certification at the plant or at an appropriate simulator as a qualification requirement for the positions of plant manager and technical manager. This certification should either be conducted or approved by the NRC staff.

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[8] The title "radiation protection manager" is used by many licensees to describe the person responsible for the radiation protection program; other titles are equally acceptable.

2.2.2 Maintenance: Education Requirements

Section 4.2.3 discusses the education requirements for the maintenance manager. The recommendation for the maintenance manager to have nondestructive testing familiarity, craft knowledge, and an understanding of electrical, pressure vessel, and piping codes and standards should be construed as a requirement.

2.2.3 Radiation Protection: Training and Experience

2.2.3.1 Minimum Formal Training. Section 4.4.4a. requires that the Radiation Protection Manager have some formal training in radiation protection. Topics to be included in this training should include as a minimum, personnel dosimetry, air sampling, shielding, radiological biology, and radiation protection instrumentation.

2.2.3.2 Certification and Experience. In general, an individual who is certified by the American Board of Health Physics in accordance with the "Power Reactor Health Physics Certification Program" dated November 1978 is considered as having met the requirements specified in Section 4.4.4 of the draft standard for the Radiation Protection Manager[8] provided that the individual has spent a minimum of 2 of the 6 years immediately preceding application for certification in a position of supervisory capacity in a health physics program for an operating nuclear power plant.

2.2.3.3 Experience Requirement. Section 4.4.4 states that the Radiation Protection Manager shall have 4 years of experience in applied radiation protection and that at least 3 years shall be in applied radiation protection work in a nuclear facility dealing with radiological problems similar to those encountered in nuclear power plants. Additionally, the 4 years of experience in applied radiation protection required by the draft standard should be professional-level experience and the 3 years of experience at a nuclear facility should include supervision of the activities of radiation protection technicians.

2.3 Shift Supervisors

2.3.1 Education Requirements

Section 4.3.1.1 discusses the requirements for qualification of shift supervisors (person in charge of shift operations). In lieu of the education requirements of the standard, the shift supervisor should have at least a Bachelor of Science degree that includes at least 60 semester hours in mathematics, reactor physics, chemistry, materials, thermodynamics, fluid mechanics, heat transfer, and electrical and reactor control theory. (A discussion of the education requirements for shift supervisor is included as Appendix A to this regulatory guide in an additional effort to obtain public input in developing this position. Specific comments on Appendix A are invited.)

2.3.2 Corporate Certification of Candidates

In order to ensure greater management involvement in the selection of shift supervisors, certification of the competency of shift supervisor candidates should be performed by the highest level of corporate management responsible for plant operations (for example, the Vice President for Operations). This certification should be documented and the documentation retained for at least as long as the individual is employed at the plant or by the organization that operates the plant.

2.4 Professional-Technical Group Leaders

2.4.1 Instrumentation and Control and Chemistry and Radiochemistry: Experience Requirements

Sections 4.4.2 and 4.4.3 contain experience requirements for the leaders of the instrumentation and control professional-technical group and chemistry and radiochemistry professional-technical group. Consistent with the experience requirements of the standard for the leader of the reactor engineering group, the experience for those individuals discussed in Sections 4.4.2 and 4.4.3 should be 4 years of professional-level experience, 2 years of which should be in instrumentation and control or chemistry and radiochemistry, respectively. Included in this experience should be 1 year of nuclear power plant experience as required by the draft standard for these positions.

2.4.2 Chemistry and Radiochemistry: Credit of Training Toward Experience

Section 4.4.3 allows 1 year of credit toward nuclear power plant experience for a chemistry and radiochemistry training program. In order to ensure that any such training program cannot be used to fulfill the requirement of the standard for the chemistry and radiochemistry group leader to have 1 year of nuclear power plant experience in radiochemistry, the equivalence of the training program to experience should be allowed only to a maximum of 6 months.

2.5 Training Coordinator: Training

Section 4.4.7.1 discusses the qualification requirements for the Training Coordinator. Item C, "Training," of this section should include the training required by Section 5.4, "General Employee Training," of the draft standard.

2.6 Simulator Instructor: Plant-Specific Senior Operator License

Section 4.4.7.2 states: "Instructors who provide instruction on the simulator shall hold a senior operator license for a similar unit (PWR, BWR, HTGR) or have been certified at an appropriate plant simulator." In lieu of this provision, training center and facility instructors who teach systems, integrated responses, transients, and simulator courses should demonstrate their competence to the NRC by successful completion of a senior operator examination applicable to the facility for which the courses are intended. (In the case of simulator instructors, the examination should be for the plant for which the simulator was constructed.) Additionally, such instructors should participate in appropriate requalification programs to ensure that they are cognizant of current operating history, problems, and changes to procedures and administrative limitations.

2.7 Shift Technical Advisor: Training

Section 4.4.8 discusses requirements for the position of Shift Technical Advisor and refers only to Section 5.3.3, "Training for Shift Technical Advisor with Bachelor Degree Without an NRC Senior Operator License," for training requirements. The implication of this section is that candidates for the Shift Technical Advisor position who are licensed senior operators would not require additional training. This is not consistent with the NRR requirements for the Shift Technical Advisor as forwarded by NRR letters to licensees of operating plants on September 13 and October 30, 1970; applicants for pending operating licenses on September 27 and November 9, 1979; and applicants for pending construction permits and licensees of plants under construction on October 10 and

November 9, 1979. The portion of these letters pertaining to the Shift Technical Advisor position has been included as Appendix B to this guide and should be followed.

2.8 Technicians: Training

Section 4.5.2 discusses training for technicians, including radiation protection technicians, and references Sections 5.3.4 and 5.4 for necessary training. In amplification of these requirements, the training program for radiation protection technicians should include the training in Sections 5.3.4 and 5.4 as well as training in the following areas:

a. Principles of radiation protection,

b. Standards and regulatory requirements concerning radiation protection,

c. The type and magnitude of potential radiological hazard for each plant system,

d. Responsibilities and authorities of their position, and

e. Tasks to be performed by the technician in normal, abnormal, and emergency conditions.

Additionally, each technician should have demonstrated an understanding of the elements of the training program by satisfactory completion of both a written examination and a practical demonstration of the tasks referred to in Item e. above.

2.9 Engineer-in-Charge: Training

Section 4.6.1 discusses requirements for the Engineer-in-Charge and states that this is normally an offsite position. Training in accordance with Section 5.4, "General Employee Training," should be provided independent of whether this is an onsite or offsite position.

3. TRAINING: EXCEPTIONS AND SUPPLEMENTS

3.1 Based on Task Analysis of Positions

Section 5, "Training," specifies training programs for positions identified in Section 4, "Qualifications," of the standard. The guidance provided in Section 5 on the content and conduct of training programs, in particular training programs for nonlicensed personnel, is very general and is of marginal value in determining the proper content of such training programs and the way the programs should be conducted. For most positions identified in the standard, it is stated that "specialized training will be based upon an analysis of the individual's background, abilities, and responsibilities and the effect his area of responsibility could have on overall plant safety." The standard requires that a task analysis (knowledge/skill necessary to perform the job) be written for most positions. The NRC staff agrees that the task analysis is a step in the right direction for the short term (see Task I.A.2.2, "Training and Qualifications of Operations Personnel," of NUREG-0660) pending development of more definitive guidance on the content and conduct of training programs.

Therefore, a position task analysis should be conducted for all operations personnel in which the tasks performed by the person in each position are defined, and the required training, in conjunction with education and experience, is identified to provide assurance that the tasks can be effectively performed. The position task analysis should include normal and emergency duties and place emphasis on the role played by every member of an operations organization that ensures safe plant operation.

3.2 Operations Personnel

3.2.1 Classroom, In-Plant, and Simulator

As part of the training and retraining program for operations personnel and in addition to the training and retraining addressed in Section 5 of the standard, instruction should be provided to operations personnel in the use of plant systems to control or mitigate accidents in which the core may be severely damaged. Such training should include classroom training, in-plant training, and simulator training.

3.2.2 Simulator and In-Plant Drills

Section 5.5 discusses retraining of operations personnel. Section 5.5.1.2.1 discusses control manipulations and plant evolutions that should be conducted as part of the requalification program for licensed operators and senior operators. Section 5.5.1.2.1 stipulates that the evolutions are to be conducted at simulators. While training conducted at simulators is a necessary part of the requalification program, the training program for operations personnel should also consist of in-plant drills. For tasks performed by shift operations personnel in response to off-normal or accident situations, licensees should ensure that sufficient in-plant drills are conducted to enable personnel to maintain proficiency in those tasks. Drills of a walk-through nature are acceptable in the short term. (Over the long term, the NRC staff will give consideration to the need for a standard dealing with in-plant drills.)

3.2.3 Qualifying Examinations

Section 5.5.1.3.1, "Annual Written Examination," states: "A grade of less than 70% in any category shall require accelerated requalification in that category. A grade of less than 75% overall requires accelerated requalification in all categories graded less than 75%." In lieu of this provision, the criteria for requiring a licensed individual to participate in accelerated requalification should be modified to be consistent with the new passing grade for issuance of a license: 80% overall and 70% for each category (i.e., a grade of less than 70% in any category should require accelerated requalification in that category. A grade of less than 80% overall should require accelerated requalification in all categories graded less than 80%).

D. IMPLEMENTATION

This proposed guide has been released to encourage public participation in its development. Except in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method to be described in the active guide reflecting public comments will be used in the evaluation of applications for construction permits and operating licenses after the implementation date to be specified in the active guide. Implementation by the staff will in no case be earlier than May 1, 1981.

Criteria equivalent to some positions of this regulatory guide have been previously applied to applicants and licensees by NRR letters to operating plant licensees on September 13, 1979, and October 30, 1979; applicants for pending operating plant licenses on September 27, 1979, and November 9, 1979; applicants for pending construction permits and licensees of plants under construction on October 10, 1979, and November 9, 1979; and to all applicants and licensees on March 28, 1980. The NRC intends to apply the remaining parts of this regulatory guide to operating plants and plants under construction in conjunction with implementation of Task I.B.1.1, "Organization and Management of Long-Term Improvements," of NUREG-0660. Accordingly, it is the staff's intention that all operating reactor licensees will be required to submit a plan by May 1981 for implementation of criteria for qualification of staff personnel that are expected to be equivalent to the remaining positions of this guide. These plans will describe both actions taken to implement previous requirements and commitments made (including schedules) and actions planned for implementing the remaining criteria. (It is expected that implementation of upgraded qualifications for shift supervisors, in particular, the requirements for a Bachelor of Science degree, will be completed as expeditiously as possible, but no later than January 1, 1986.)

REFERENCES

1. Part 50, "Domestic Licensing of Production and Utilization Facilities," Title 10, Code of Federal Regulations, U.S. Government Printing Office (GPO), Washington, D.C. 20402.
2. Part 55, "Operators' Licenses," Title 10, Code of Federal Regulations, U.S. Government Printing Office, Washington, D.C. 20402.
3. December 6, 1979, Draft Standard ANS 3.1, "Qualification and Training of Personnel for Nuclear Power Plants," American Nuclear Society, 555 North Kensington Avenue, LaGrange Park, Illinois 60525.
4. "Report of the President's Commission on the Accident at Three Mile Island" (Library of Congress Catalog Card Number 79-25694), U.S. Government Printing Office, Washington, D.C. 20402 or Pergamon Press, Inc., Maxwell House, Elmsford, NY 10523.
5. NUREG/CR-1250,* "Three Mile Island, A Report to the Commission and to the Public," Volumes I and II, U.S. Nuclear Regulatory Commission, (Vol. I) January 1980 and (Vol. II) May 1980.
6. NUREG-0616, "Report of Special Review Group, Office of Inspection and Enforcement, on Lessons Learned from Three Mile Island," U.S. Nuclear Regulatory Commission, December 1979.
7. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," U.S. Nuclear Regulatory Commission, July 1979.
8. NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report," U.S. Nuclear Regulatory Commission, August 1979.
9. NUREG-0645, "Final Report of Bulletins and Orders Task Force of the Office of Nuclear Reactor Regulation," Vols. 1 and 2, U.S. Nuclear Regulatory Commission, January 1980.
10. ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel," American Nuclear Society.
11. ANSI/ANS 3.1-1978, "Selection and Training of Nuclear Power Plant Personnel," American Nuclear Society.
12. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," GPO Sales Program, Distribution Services Section, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, May 1980.

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*NUREG series documents and reports are available at current rates through the GPO Sales Program, Distribution Services Section, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 or from the National Technical Information Service, Springfield, Virginia 22161.

13. Commission Paper SECY 79-330E,** "Qualifications of Reactor Operators," U.S. Nuclear Regulatory Commission, July 30, 1979.

14. Office of Nuclear Reactor Regulation, Harold R. Denton Letter to All Power Reactor Applicants and Licensees, "Qualifications of Reactor Operators," March 28, 1980.**

15. "Utility Management and Technical Resources," Executive Summary and Vols. I, II, and III, Teknekron Research, Inc., 1483 Chain Bridge Road, McLean, VA 22101, May 1980.

16. February 25, 1980, Draft Document, "Draft Criteria for Utility Management and Technical Competence," Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission.**

17. Commission Paper SECY-80-242, "Independent Safety Engineering Group," U.S. Nuclear Regulatory Commisson, May 13, 1980.

18. Commission Paper SECY-80-243, "Shift Technical Advisors," U.S. Nuclear Regulatory Commission, May 13, 1980.

19. NUREG/CR-1280, BETA-103, "Power Plant Staffing," U.S. Nuclear Regulatory Commission, Washington, D.C., March 1980.

**Commission Papers with SECY designators and unclassified NRC correspondence are available for inspection or copying for a fee in the NRC Public Document Room, 1717 H Street NW., Washington, D.C.

APPENDIX A

EDUCATION REQUIREMENTS FOR SHIFT SUPERVISORS

1. PROPOSED REGULATORY POSITION 2.3.1

Regulatory Position 2.3.1 states that in lieu of the education requirements specified in Section 4.3.1.1 of the December 6, 1979, Draft Standard ANS 3.1 for qualification of shift supervisors (person in charge of operations on a particular shift), the shift supervisor should have at least a Bachelor of Science degree that includes at least 60 semester hours in mathematics, reactor physics, chemistry, materials, thermodynamics, fluid mechanics, heat transfer, and electrical and reactor control theory.

This regulatory position is responsive to Commission direction of November 27, 1979. The Commission in its approval, subject to certain modifications, of staff recommendations in Commission Paper SECY 79-330E and a supplemental paper, SECY 79-330F, "Qualifications of (Power) Reactor Operators," stated that all Commissioners support increasing qualifications of applicants, and, as an initial step, the staff's recommendation on experience requirements for applicants for senior reactor operator licenses is accepted. The Commission requested that the staff prepare for the Commission's consideration proposed rules that would be consistent with the recommendations of Item 1.6 on shift supervisor and senior reactor operator qualifications of the Lessons Learned Task Force (NUREG-0585).

The Lessons Learned Task Force in NUREG-0585 provided the following recommendations regarding shift supervisors:

"A program for raising the qualification requirements for shift supervisors and senior reactor operators should be established. The distinction being made in present practice between senior reactor operators (e.g., shift foreman in a multi-unit station) and shift supervisors should be recognized. As a short-term action pursuant to NUREG-0578 (until such time as staffing and qualification of shift personnel and the control room man-machine interface requirements are upgraded), each licensee has been required to provide an on-shift technical advisor to the shift supervisor. Within the next five years, it is recommended that the qualifications of senior reactor operators and shift supervisors be upgraded as indicated below. Qualification requirements for applicants for licensing prior to initial fuel loading may require special additional considerations, particularly with respect to experience.

"(1) Shift Supervisor (person in charge of operations on shift at the station) - Shift Supervisors should have at least a Bachelor of Science degree or equivalent training and experience in engineering or the related physical sciences. The Shift Supervisor should also hold a senior reactor operator's license (issued under new proposed requirements defined below) and have served as a reactor operator for one year or senior reactor operator for six months. In establishing equivalency with a Bachelor of Science degree, consideration should be given not only to formal courses in engineering and related sciences, but also to education in the liberal arts. It is recommended that the use of the equivalency to a Bachelor of Science degree be exercised to only a limited degree and that most shift supervisors hold degrees. It is also recommended that shift supervisor qualifications include leadership training and experience.

"(2) Senior Reactor Operator (e.g., shift foreman in a multi-unit station) - Senior Reactor Operators should have at least the same general technical education and specific training in transient and accident response characteristics of nuclear power plants as recently articulated for the shift technical advisor. Additional recommendations for upgrading senior reactor operator qualifications are identified in the Commission Paper SECY 79-330E on Qualification of Reactor Operators."

In comparing proposed Regulatory Position 2.3.1 and the recommendations of the Lessons Learned Task Force it should be noted that the Lessons Learned Task Force recommended (1) that shift supervisors have at least a Bachelor of Science degree or equivalent training and experience in engineering or the related physical sciences and (2) that the shift supervisor also hold a senior reactor operator's license issued under new requirements proposed by the Lessons Learned Task Force (senior reactor operators should have at least the same general technical education and specific training in transient and accident response characteristics of nuclear power plants as recently articulated for the shift technical advisor). The provision of this regulatory position for 60 semester hours of technical education is derived from the general technical education requirements for shift technical advisors (see Appendix B to this guide).

Proposed Regulatory Position 2.3.1 is more stringent than the position the Advisory Committee on Reactor Safeguards (ACRS) may deem necessary. The Advisory Committee on Reactor Safeguards in its report of December 13, 1979,* to Chairman Ahearne on the Lessons Learned Task Force Final Report (NUREG-0585) stated that it gave general support to the recommendations related to personnel qualification and training; however, the ACRS also provided the following comment concerning qualifications of shift supervisors.

"The ACRS believes that, although a broader technical background should be required of Shift Supervisors, it may be neither necessary nor practical to require that all Shift Supervisors have a Bachelor of Science Degree. The Committee recommends that the NRC define its criteria for 'equivalent training and experience in engineering or the related physical sciences.' The ACRS believes that a training program tailored to the requirements of reactor operation, possibly of less than four years duration, may provide a practical alternative to a formal degree program. The Committee believes that the NRC should define the scope and duration of a training program that may be considered as an acceptable alternative to a degree curriculum."

The differences between the comment of the ACRS and proposed Regulatory Position 2.3.1 will be discussed in Section 3 of this appendix when addressing the alternative education requirements for shift supervisor that could be considered. Prior to that discussion, recommendations or comments of other groups on qualification requirements for shift supervisors are presented.

2. RECOMMENDATIONS AND COMMENTS FROM OTHER GROUPS

The Report of the NRC Special Inquiry Group (NUREG/CR-1250) recommended that the NRC should require every licensee to hire a cadre of graduate engineers knowledgeable in reactor engineering and physics and that each engineer be provided with training in specific characteristics of the plant, with special emphasis on integrated plant response and transient behavior. The Special Inquiry Group further recommended that the utility should be required to deploy on every shift at least one such engineer as supervisor whose qualifications as shift manager, not as an "advisor," have been examined by the NRC.

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* Document is available for inspection or copying for a fee in the NRC Public Document Room at 1717 H Street NW., Washington, D.C.

Basic Energy Technology Associates, Inc. (BETA), in its report to the NRC on power plant staffing (NUREG/CR-1280) discusses the differences in NRC, commercial, and Naval practice for shift supervisors. Attachment 1 to this appendix is an excerpt from the BETA report concerning shift supervisor qualifications. The BETA report recommends that a new position entitled "Shift Engineer" be created. The "Shift Engineer" would be a degreed engineer who would normally function within the technical organization but is assigned to the Operations Manager to provide shift engineering coverage. The BETA report also indicates that the "Shift Engineer" should have the power and responsibility to direct the shift supervisor in the event of an emergency.

If the "Shift Engineer" has the power and responsibility to direct the shift supervisor and is licensed as a senior reactor operator with extensive operating experience, then the difference between the BETA recommendation and the proposed regulatory position is more related to organizational requirements than to qualification requirements of the "person in charge of operations on shift."

Teknekron Research, Inc., in its report to the NRC on utility management and technical resources recommended the shift supervisor have a bachelor's degree in engineering. Attachment 2 to this appendix is an excerpt from the Teknekron report concerning shift supervisor qualifications.

While the above recommendations, for the most part, have been supportive of proposed Regulatory Position 2.3.1, the Atomic Industrial Forum (AIF) had reservations about such an approach. In the February 1980 Report to the AIF Policy Committee on Follow-Up to the Three Mile Island Accident by the AIF Working Group on Action Plan Priorities and Resources, it was stated: "The requirements for a B.S. degree could have adverse effects on plant safety in that it probably would result in a higher turnover rate for these positions, thus reducing experience in this position at most plants industry wide. The turnover is expected because the person is likely to consider himself over qualified for the usual daily operations; he would not be gaining professional satisfaction. There would be no risk reduction attributable to the degree per se because it is assumed that he has sufficient fundamental education in the proper engineering disciplines. If this requirement is implemented, risks from plant operation will increase due to personnel turnover. It is suggested this item be reviewed with a consideration toward increasing requirements for fundamental education, but not requiring a degree."

Proposed qualification requirements for the shift supervisor are also included in the December 6, 1979, Draft Standard ANS 3.1, "Standard for Qualification and Training of Personnel for Nuclear Power Plants." This draft revision of ANS 3.1 states, in Section 4.3.1, that the educational requirements for this position are a high school diploma plus the equivalence of 60 semester hours of college-level education in specified technical topics.

3. ALTERNATIVE EDUCATION REQUIREMENTS

A wide variety of opinion exists concerning education requirements for shift supervisors. A discussion of proposals for education requirements for shift supervisors is provided below in an attempt to define the issues involved and to foster public comment on these issues. While the requirements related to nuclear power plant experience are not addressed in the following discussion, the importance of extensive in-plant experience for shift supervisors cannot be over emphasized.

Alternative education requirements for shift supervisors include the following:

Alternative 1: Require as a minimum a high school education.

Alternative 2: Require a high school education plus a specified number of college-level technical courses.

Alternative 3: Require a high school education plus a specified number of college-level courses in technical subjects as well as courses in humanities and social studies such as written and oral communication, applied psychology, political science, and economics.

Alternative 4: Require a Bachelor of Science degree in engineering or a related physical science.

Alternative 5: Require a Bachelor of Science degree in engineering or a related physical science that includes a specified number of courses in technical subjects as well as courses in humanities and social studies such as written and oral communication, applied psychology, political science, and economics.

The following discussion explains why Alternatives 1, 2, and 4 are deemed unacceptable and compares the acceptability of Alternatives 3 and 5.

Alternative 1 (high school diploma) is unacceptable. In particular, the technical complexity of supervising the operation of a nuclear power plant requires an education exceeding that demonstrated by a high school diploma. The demands of present practice for operator and senior operator training exceed this requirement (i.e., while a shift supervisor may have only a high school diploma, completion of the senior operator training program requires education beyond the high school level.)

Alternative 2 (high school diploma plus completion of a specified number of college-level technical courses) is also unacceptable. While this alternative qualifies a supervisor to deal with technical areas, it does not provide a broad-based education in nontechnical subjects such as management, leadership, and written communication that is necessary to deal with many of the nontechnical responsibilities of the shift supervisor, particularly, supervision of plant personnel.

Alternative 4 (Bachelor of Science degree in engineering or a related physical science) is also unacceptable. There are some Bachelor of Science (B.S.) degree programs that do not meet the education requirements for a shift supervisor. Although an engineer holding a B.S. degree would have completed many of the subjects considered necessary to qualify as a shift supervisor, having successfully earned a degree in engineering does not ensure knowledge of such specific areas as fluid mechanics and reactor control theory that is necessary to the shift supervisor.

The remaining discussion will be limited to a discussion of Alternatives 3 and 5. Proposed Regulatory Position 2.3.1 (Alternative 5) is selected as the optimum choice because it quantifies the required technical courses as 60 semester hours in mathematics, reactor physics, chemistry, materials, thermodynamics, fluid mechanics, heat transfer, and electrical and reactor control theory. Alternative 3 does not quantify the number of technical courses required.

When considering technical courses, two questions are readily identifiable. First, is the number of courses and the area of their coverage sufficient? Since the number of technical courses required for a Shift Supervisor in Alternative 5 is the same as that required for a shift technical advisor (Regulatory Position 2.7), the course requirement appears to be acceptable. Second, should the requirement stipulate that the courses be taught at an accredited college or university? The level of instruction of the courses is required to be beyond the high school level. Therefore, the only practical way, pending accreditation of technical training institutions other than 2- or 4-year accredited colleges or universities, to specify the required level of instruction is to provide that such instruction be satisfactorily completed at or conducted by an accredited college or university.

The technical education requirements of Alternatives 3 and 5 are consistent, but the nontechnical education requirements of Alternatives 3 and 5 differ. It is expected that the nontechnical liberal arts curriculum requirement of Alternative 3 would consist of a minimum of 20 semester hours conducted at or by an accredited college or university in such courses as psychology, management, leadership, sociology, and other liberal arts courses thereby providing a broad-based education. This broad-based education would equip a shift supervisor to deal with many of the nontechnical responsibilities of the position such as communicating with shift personnel and providing leadership for shift activities. Completion of the required 60 hours of technical courses and estimated 20 hours of nontechnical courses amounts to 80 semester hours or approximately 75% of the work necessary to obtain a degree. Completion of only an additional 40 to 50 semester hours would be required to obtain a college degree. It is expected that an individual who completes 75% of the work necessary to obtain a degree would likely complete the remainder. Therefore, realistically, there may be little difference between Alternatives 3 and 5.

A college degree indicates a level of accomplishment and self-discipline that helps produce self-confidence in an individual and helps establish leadership qualities necessary to a shift supervisor. The wide exposure of the individual to both technical and liberal arts courses provides a more well-rounded education than exposure to only technical courses that could improve a shift supervisor's ability to direct shift operations activities.

A degree is based on a well-thought-out curriculum with required courses integrated and dovetailed to complement one another in a consistent manner in order to equip a person to consider problems and make decisions in a constructive way in a particular field. Without such planning, 60 hours, or any other required number of courses, is meaningless--it could be a hopeless hodgepodge of unintegrated and only vaguely related information. Therefore, Alternative 5 with its degree requirement is probably the best choice.

4. IMPLEMENTATION

The Lessons Learned Task Force in its final report (NUREG-0585) recommended a phased program to take place within the next 5 years for upgrading the qualifications of shift supervisors and senior reactor operators. Whether training of present personnel is upgraded or degreed engineers are introduced into the training program, it is expected that it would take a minimum of 5 years to fully respond to the upgraded requirements stated in Alternatives 3 or 5.

A program of upgrading the qualifications of individuals now employed as shift supervisors at operating plants should be initiated by the utilities to ensure that these individuals possess the necessary technical competence and management skills to adequately perform their jobs. Due credit should be given shift supervisors for demonstrated ability to perform their duties. The qualifications of current shift supervisors and the utility's program for upgrading qualifications for shift supervisors should be reviewed on a case-by-case basis by the Office of Nuclear Reactor Regulation at the time of renewal of the shift supervisor's senior operator license.

In implementation of the regulatory position on shift supervisors, consideration will need to be given to the AIF concern that such a position may cause a high turnover rate and thereby adversely affect plant safety.

EXCERPT FROM NUREG/CR-1280, "POWER PLANT STAFFING"

VI. SHIFT SUPERVISORS

A. Definitions

In the context of this review, shift supervisors will be considered licensed senior reactor operators (SRO's) and will be compared with Engineering Officers of the Watch (EOOW's).

An Engineering Officer of the Watch (EOOW), insofar as a nuclear ship is concerned, is an officer who has been selected, trained, qualified, and designated as a nuclear trained officer. He has been qualified as an EOOW in his ship by his Commanding Officer. He is the senior officer on watch in the engineering plant of the ship. All persons on watch in the engineering plant report to him. He is responsible for the operation of the plant, its safety, emergency action and anything going on in the plant. He directs all operations.

B. NRC Requirements for Shift Supervisors

NRC requirements for eligibility, training, qualification and requalification of shift supervisors are contained in ANSI N18.1-1971 and NUREG-0094. These requirements are summarized in Table 1 of this report.

C. Industry Practices for Shift Supervisors

As in the case of operators, civilian industry practices generally follow the NRC requirements. However, as previously indicated, ANS 3.1-1978 and proposed Revision 2 to Regulatory Guide 1.8 are used even though not required. All of the comments made in Section V.B of this report concerning industry problems with operators equally apply to the case of shift supervisors.

D. Navy Practices

Before a naval officer can qualify as an Engineering Officer of the Watch (first-line operating supervisor) on a nuclear ship he will have met the following requirements:

1. He will have entered into the Navy's nuclear power program by applying (volunteering) and meeting the following requirements:

  a. Age requirements: no older than 27 years of age.

  b. College graduate (4 year curriculum) having successfully completed one year of calculus through differential and integral calculus, and one year of calculus-based physics.

  c. Physically qualified.

  d. Meet requirements of moral turpitude sufficient to be granted an appropriate security clearance.

  e. Have been interviewed in Washington, D.C. headquarters of the Department of Energy's Deputy Assistant Secretary for Naval Reactors. These interviews consist of at least three individual interviews by senior technical staff personnel and the Deputy Assistant Secretary himself. In about two-thirds of the cases, written examinations in math and physics are administered during this interview period.

2. The Navy acquires its officers for the nuclear program through three sources: U.S. Naval Academy, NROTC colleges, Nuclear Power Officer Candidate (NUPOC) program. Officers selected for the nuclear program from the NUPOC program must attend the Navy's Officer Candidate School (OCS) for 16 weeks.

3. Successful completion of the Navy's 6 months Nuclear Power School (NPS). This course teaches basic theory relating to nuclear power at a higher level than that taught to enlisted personnel. All instruction is conducted in the classroom.

4. Successful completion of a 6 months practical course of instruction at one of the Navy's 8 nuclear prototype plants. Students will have actually "qualified" on the plant. During this phase, the officer will qualify on all enlisted watch stations in addition to qualifying as an EOOW.

5. Will have been assigned to a nuclear powered ship and will have "qualified" on that ship's plant. This takes about 9 months and involves the following:

  a. A basic engineering qualification (BEQ) course which, in addition to being a review of course material covered at NPS, covers basic reactor plant theory and application to the specific plant installed on his ship.

b. A watch qualification program involving standing watches under instruction throughout the plant.

c. The watch qualification program also involves completion of each item of a watch qualification guide by obtaining signatures, usually several hundred, that he has demonstrated, through discussion, actual practice, or written tests his knowledge and ability.

d. He will be given a final comprehensive written examination and a series of oral examinations. He must be given his final oral examination by the ship's Commanding Officer.

6. This entire program, from the time he enters the Navy as an officer until he stands his first EOOW as a qualified watch stander is about 2 years.

E. Differences and Recommendations for Shift Supervisors

1. Difference:

EOOW's are naval officers selected into the program through a tough but well-defined system. Quality input is maintained even to the sacrifice of quantity. High standards of performance are instilled from the first moment of selection and are emphasized throughout the training program. Each candidate knows he has a 5 year commitment. He may fail, but he can not quit. The training program is structured so that all students must work hard to succeed. No one, regardless of his background or intelligence, can just breeze through. Standards for passing or failing the courses are clearly defined and enforced without waivers. There is every incentive to finish the courses and finish them well. There is no incentive to "drift along". He is constantly examined as to his understanding and retention of knowledge. His examinations are orals or essay written. There are not true or false or multiple choice examinations. There is no self-pace teaching. His training is competitive and he knows it. The higher his relative standing, the better chance he has of selecting his duty station. His prototype instructors are qualified and experienced operators, many of whom are sea-returnees. His rewards are ample:

a. Recognition in the form of special bonuses which are substantial.

b. Pride in being part of a small, elite group of officers who have successfully passed through the most difficult program the Navy has to offer.

c. A sure path to future, better-than-average promotion if he continues satisfactory performance.

d. The prospect of a select civilian career if he elects to resign at the end of his commitment.

e. Knowledge that he is an integral part of the nation's number one major deterrent to war.

In the civilian nuclear industry a shift supervisor comes from two sources. He can either be promoted up from the ranks of an operator, or he can be brought in directly from outside and made a shift supervisor without passing through the job as an operator. In either case, his selection, training and qualification generally follow that of an operator but with greater experience required. All of the comments provided in Section V of this report relating to differences and recommendations have direct applicability to shift supervisors.

However, there are two additional differences that should be highlighted which are unique to the shift supervisor and the EOOW.

The one difference has been discussed in the various reports and studies emanating from the Three Mile Island accident and has to do with the proper manning of the control room. NRC has already issued interim requirements on the stationing of a Shift Technical Advisor in the Control Room.

Recommendations:

Our recommendation is to create a new position entitled "Shift Engineer". He would be a degreed engineer who would normally function within the technical organization but is assigned to the Operations Manager to provide shift engineering coverage. This position is created for the following purposes and reasons:

a. If it is assumed that the requirements for becoming a shift supervisor (senior reactor operator) remain such that he need not be an experienced engineer (college graduate type), then there exists the need for such a person on shift who can make engineering judgments. This would be the function of the Shift Engineer.

b. The possibility exists to change the requirements for a shift supervisor such that he must be a college graduate engineer. This alternative was not selected because it would close off an advancement path for reactor operators. While some may consider this to be a minor issue, the reviewers, based on their Navy experience, do not. The civilian nuclear power industry must be able to provide an attractive career path for reactor operators or else face the prospect of heavy turnover or lower quality applicants.

c. There is also the suggestion that the position of Shift Engineer be filled only when a shift supervisor is not an engineer, or that the Shift Engineer position be an interim measure until such time as all shift supervisors meet the engineer eligibility requirements or their equivalent. We do not agree with this approach. Regardless of whether or not a shift supervisor is an engineer, there should always be present in the control room an engineer whose primary interest, background and experience is technical in nature.

d. The functions of the Shift Engineer would be as follows:

(1) He acts as a technically qualified observer to plant operations.

(2) He has the power to order the plant put into a safe condition in the event of an emergency.

(3) He does not report to the Shift Supervisor--he is an independent observer similar to the NRC inspector on shift. However, he has the power and responsibility to direct the Shift Supervisor in the event of an emergency or accident.

(4) He has the wherewithal to contact appropriate technical personnel to obtain technical assistance, thus allowing the Shift Supervisor to focus his attention on plant operation.

(5) If, during the course of normal operations, it is discovered that a given procedure requires modification, the Shift Engineer has the responsibility to resolve the problem, correct the procedures in accordance with approved methods, and to provide the results to the Shift Supervisor for his accomplishment.

e. The Shift Engineer would be a licensed Senior Reactor Operator and will have had operating experience as outlined in Table 2 of this report.

EXCERPT FROM VOLUME I, "UTILITY MANAGEMENT AND TECHNICAL RESOURCES," OF THE TEKNEKRON RESEARCH, INC., REPORT

1.5 ONSITE MANAGEMENT RESOURCES

One man must be in charge of the plant at all times. Since in the normal work-week, the Plant Manager can be expected to be onsite less than 25% of the time, the Shift Supervisor has short-term onsite management responsibility (in the absence of the Assistant Plant Manager). (The accession of management responsibility by the Plant Manager or his designee in the event that the long-term resources should be called upon is discussed in Section 2.3.)

The Shift Supervisor is the single most important resource in the event of an accident. Not only is he solely responsible for any hands-on operations performed on the system, but he must also be the "intellectual leader" of the team. He must be able to evaluate and act upon the technical input supplied by individuals on his onsite team and by his offsite resources. He must also be able to communicate essential information to the long-term resources if and when they arrive onsite. It is essential that he be thoroughly familiar with the plant, that he be exceedingly well trained (and retrained), and that he command the respect of his colleagues. Although this is a stringent requirement, it is not possible to overemphasize the importance of this individual in an accident situation.

With the foregoing in mind, and recognizing that the Shift Supervisor is equivalent to the Plant Manager when the latter is absent, we recommend the following minimum qualifications for the Shift Supervisor:

1. A bachelor's degree in engineering
2. Two years of operating experience at a nuclear power plant
3. One year of experience at the specific nuclear plant to be operated (which can be satisfied by pre-operational testing)
4. An NRC Senior Operator's License
5. Four years of supervisory experience

The nuclear plant experience satisfies one of the criteria under the collective technical qualifications given in Table 3. We feel strongly that Shift

Supervisors should have this operating experience "under their belt." Additionally, the Shift Supervisor should have some leadership experience. A quantification of this experience is judgmental. We have adopted the four years of supervisory experience qualification from the recommendations of the ANSI Committee (ANS-3.1) for the Plant Manager.

The degree requirement (B.S. in Engineering) is a deviation from the collective qualifications of the onsite technical resources given in Table 3. Although we are committed to the recommended collective technical resources presented earlier, we also support this requirement in the category of management resources. In our judgment, the degree will provide the Shift Supervisor with additional depth, maturity, problem solving abilities, and respect that cannot be equated to simple technical requirements.

The degree will also provide the Shift Supervisor with the basic equivalent course work in many of the areas of expertise cited earlier, as well as the foundation to master most of the others. As a final note, we strongly recommend that the Shift Supervisor possess the Transient Analysis experience discussed earlier and in Chapter 4.

APPENDIX B

CRITERIA FOR SHIFT TECHNICAL ADVISOR
(AN EXCERPT FROM NRR SEPTEMBER 13, 1979, LETTER TO ALL OPERATING POWER PLANTS)

In developing the recommendation for the Shift Technical Advisor, the Lessons Learned Task Force concentrated on the two functions that needed to be provided, namely, an accident assessment function and an operating experience assessment function.[1] The proper performance of these functions requires the provision of certain characteristics described in the following paragraphs.

A. Accident Assessment Function

1. General Technical Education

The technical education of at least one person in the control room under off normal conditions should include basic subjects in engineering and science. The purpose of this education is to aid the operator in assessing unusual situations not explicitly covered in the current operator training. The following is a tentative list of areas of knowledge that are considered to be desirable:

Mathematics, including elementary calculus
Reactor physics, chemistry and materials
Reactor thermodynamics, fluid mechanics, and heat transfer
Electrical engineering, including reactor control theory

These areas of knowledge should be taught at the college level and would be equivalent to about 60 semester hours. Although a college graduate engineer would have many of these subjects and more that would not be essential, some engineers might be deficient in a few of these specific areas, e.g., reactor physics. Although the time to teach these subjects to a licensed senior reactor operator could be as short as two years, depending on the scope and content of the subjects, the selection of a graduate engineer would likely be a more rapid means of fulfilling this characteristic.

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[1] The staff has accepted the assignment of these two functions to two separate groups at the prerogative of individual licensees.

2. Reactor Operations Training

All persons assigned to duties in the control room should be trained in the details of the design, function, arrangement and operation of the plant systems. This training is necessary to assure that the meaning and significance of instrument readings and the effect of control actions are known. A licensed operator or supervisor of an operator would not be required to have further training in order to fulfill this characteristic. A graduate engineer not previously licensed or trained as an operator or senior operator would require additional training in order to fulfill this characteristic.

3. Transient and Accident Response Training

In addition to the training in normal operations, anticipated transients, and accidents presently required of operators and senior operators, one person in the control room under off normal conditions should be trained to recognize and react to a wide range of unusual situations including multiple equipment failures and operator errors. This training should not be limited to written procedures or specific accident scenarios, but should include the recognition of symptoms of accident conditions such as complex transient responses or inadequate core cooling and possible corrective actions. The purpose of this training is to broaden the ability for prompt recognition of and response to unusual events, not to modify the instinctive, rapid procedural response to transients and accidents provided by reactor operators. The training is required in recognition of the fact that real accidents inherently are initiated and accompanied by unusual and unexpected events. The training is also to emphasize need to focus on the essential parameters that indicate the status of the core and the primary coolant boundary. This additional training would take up to a year to accomplish for a person not already experienced in nuclear plant transient and accident analysis or evaluation. Both inexperienced graduate engineers and currently licensed operators would require additional training to fulfill this characteristic.

4. Detachment from Operations

The plant response assessment function requires a measure of detachment from the manipulation of controls or immediate supervision of operators. This is intended to provide the perspective and the time for assessing plant conditions and advising on appropriate operator actions. It has been called a safety monitor characteristic. Currently only three operators would normally be in the control room at the time an unusual event occurred, and it is allowed that at times there would be fewer. This number is only enough to satisfy the demands for prompt control and supervisory actions under off normal conditions. The time necessary to make a considered assessment and permit independent monitoring of plant safety requires one more person in the form of the Shift Technical Advisor or some alternative in the control room.

5. Independence from Operations

In order to provide both perspective in assessment of plant conditions and dedication to the safety of the plant, this function should have a clear measure of independence from duties associated with the commercial operation of the plant. In an accident situation where command authority should not be diluted, complete independence is not desirable and is not necessary to the safety assessment function.

6. Availability

This capability should be readily available in the control room, preferably immediately at all times, but at most within ten minutes. Having this capability on duty for each shift is the best approach.

B. Operating Experience Assessment Function

1. Independence from Operations

A measure of independence is required to provide for effective safety monitoring of operating experience at the individual plant and at plants of like design. The assessment of operating experience at the assigned plant and other similar plants and the routine monitoring of the safety of plant operations is usually compatible with and necessary for efficient operations. However, the demands of commercial operation can sometimes distract from or appear to override safety judgments. An independent monitoring of the safety of plant operations is intended to counter-balance the immediate and pressing needs of commercial operation.

2. Dedication

Personnel should be dedicated to the function of safety monitoring of operating experience as their primary responsibility and duty. Although reactor operating personnel have a commitment to safety that derives from self interest as well as regulatory requirements, it is only one of two primary responsibilities, the other being the continuous production of power. The assignment of safety evaluation of operating experience as a primary responsibility for certain specified individuals will reduce potential conflicts and assure adequate time to discharge the duties.

3. Diversity of Technical Knowledge

The technical knowledge of those assessing operating experience should be diverse and encompass all technical areas important to safety. The types of problems that can affect safety include all areas related to the design and operation of nuclear power plants; e.g., mechanical, electrical and fluid systems and reactor physics, chemistry and metallurgy. Recognition and understanding of a problem and its significance requires some knowledge in the relevant technical specialities and cannot depend solely on the descriptions and judgments of the persons identifying and reporting the problem. Because of the broad scope of possible technical areas and the possible interactions of components, equipment and systems, the people engaged in operating experience review should have experience in areas usually designated as systems engineering. They should also be graduate engineers, or equivalent. In addition, because of the importance of operator actions in the safety of plant operations, familiarity with or routine access to persons with the principles of human engineering or human factors should be provided.

DRAFT VALUE/IMPACT STATEMENT*

1. THE PROPOSED ACTION

1.1 Description

The applicant for a license to operate a nuclear power plant is required by the Commission's regulation to include information concerning organizational structure, personnel qualification, and related matters. The proposed action will provide updated guidance for the qualification and training of some nuclear power plant personnel.

1.2 Need

In the aftermath of the Three Mile Island nuclear plant accident, the industry, the NRC, and others have conducted a number of studies and investigations, and, as a result of their findings, have recommended changes in the numbers of nuclear power plant personnel now employed, and their qualification and organization. The principal studies have been conducted by the President's Commission on Three Mile Island, the NRC Special Inquiry Group, the Office of Inspection and Enforcement Special Review Group, the Office of Nuclear Reactor Regulation's (NRR) Lessons Learned Task Force and NRR's Bulletins and Orders Task Force. Collectively, these studies have called for a general upgrading of utility capabilities for handling routine plant operations and for coping with unusual conditions. As might be expected, the recommendations of these diverse groups are not completely compatible; what is clear, however, is that all of these groups have called for upgrading in at least two identifiable areas: management oversight and technical competence.

NUREG-0660, "NRC Action Plan Developed As a Result of the TMI-2 Accident," in Chapter I, "Operational Safety," describes actions intended to substantially improve and emphasize operational safety, an area that has not previously been given the same regulatory emphasis as nuclear power plant design. The actions have two complementary objectives: (1) to reduce challenges to the safety of the plant, and (2) to ensure proper reactions to challenges that do occur. The reduction of challenges requires a highly qualified staff that devotes unflagging attention to the proper operation of the plant, continuous monitoring to verify that plant operations are correctly performed, and correcting and improving operations based on feedback of operating experience to appropriate personnel. The proper reaction to challenges to the safety of the plant requires a thorough understanding of plant design and plant response to unusual conditions, as well as training in the diagnosis of these conditions and reaction to them.

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* Unless otherwise indicated, consult the list of references for the guide for complete information and availability of documents discussed in this draft value/impact statement.

Guidance on qualification and training of power plant operations personnel is contained in Regulatory Guide 1.8, "Personnel Selection and Training," that endorses with certain exceptions and supplements ANSI/ANS Standard 3.1. The development of both the guide and the standard it endorses are discussed below.

Changes needed in Regulatory Guide 1.8 are discussed in two items of the NRC Action Plan (NUREG-0660): Item I.A.2.6, "Long-Term Upgrading of Training and Qualifications," and Item I.B.1.1, "Organization and Management of Long-Term Improvements." Short-term actions related to personnel qualifications have already been taken by the NRC and are beyond the scope of the positions of the active version of Regulatory Guide 1.8, Revision 1-R reissued May 1977. Short-term actions previously required of applicants and licensees as well as longer-term actions relating to the overall question of upgrading personnel qualifications are documented in NUREG-0660. The ANSI/ANS Standard 3.1, "Qualification and Training of Nuclear Power Plant Personnel," (see discussion below) is also undergoing revision.

Based on existing knowledge and experience in 1971, Subcommittee ANS-3, Reactor Operations, of the American Nuclear Society Standards Committee developed a standard containing criteria for the qualification and training of nuclear power plant personnel. This standard was approved by the American National Standards Institute (ANSI) Committee N18, Design Criteria for Nuclear Power Plants, and designated ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel." Regulatory Guide 1.8, "Personnel Selection and Training," endorsing ANSI Standard N18.1-1971 was issued in March 1971. A revision to this standard was subsequently approved by the ANSI Board of Standards Review and designated ANSI/ANS 3.1-1978, "Selection and Training of Nuclear Power Plant Personnel."

In February 1979, a draft Revision 2 to Regulatory Guide 1.8 (RS 807-5) endorsing ANSI/ANS 3.1-1978 was issued for public comment. In May 1979, additional comments on the area of personnel qualifications were requested in light of experience gained from the TMI-2 accident. Copies of all comments received both on the February 1979 draft guide and as a result of the May 1979 request for additional comments were provided to the ANS-3 Subcommittee for their use during development of a revision to the 1978 standard. The draft standard incorporating upgraded requirements was approved by the ANS-3 Subcommittee on December 6, 1979. Draft Standard ANS 3.1 has been extensively revised in most areas on which comments were received.

The significant changes incorporated into the draft revised standard include (a) a reformulation of Section 4, "Qualifications," to separate the topics of education, experience, and training in order to more clearly define qualification requirements; (b) an upgrading of a number of specific qualification requirements, particularly in the area of education and experience requirements; (c) more definitive guidance on training and retraining programs including requirements for the use of position task analyses to define required training and the use of simulators in training and retraining programs; (d) a listing of additional specific control manipulations to be performed during the retraining program; (e) requirements for corporate management certification of individuals prior to licensing examinations by the NRC, and (f) qualification requirements for individuals directing preoperational and startup tests.

NRC guidance should be updated to reflect the above actions and to provide for upgrading the training and qualification of nuclear power plant personnel.

1.3 Value/Impact

1.3.1 NRC

Guidance on selection and training of personnel now contained in Regulatory Guide 1.8, Revision 1-R reissued May 1977, has been superseded by staff actions as described in NUREG-0660 and changes to the ANSI/ANS 3.1 standard.

Value

The proposed action will provide guidance consistent with upgraded qualification requirements for plant staff personnel. The proposed action will also serve to consolidate various recommendations concerning personnel qualifications for ongoing evaluation.

Impact

The impact on the NRC staff will be that of reviewing applicants' qualifications against the upgraded requirements. In some areas, this will represent an increase in the scope of review, for example, the area of education, with a consequent requirement for increased staff to perform such reviews.

1.3.2 Other Government Agencies

The proposed action will not affect other government agencies, unless the government agency is an applicant, such as TVA.

1.3.3 Industry

Value

The proposed action will provide guidance to the industry and thus reduce uncertainty as to what the NRC staff considers acceptable in the area of personnel qualification and training.

Impact

Since the proposed action represents upgraded qualification of nuclear power plant personnel in a number of areas, more highly qualified personnel may demand higher salaries from industry. Additionally, even though higher salaries are offered, there may be more positions available than qualified personnel to fill them.

1.3.4 Public

Value

The proposed action will ensure that individuals with, in most cases, increased training and qualifications operate nuclear power plants, thus decreasing the risks caused by human error to the public health and safety.

Impact

The impact to the public of the proposed action would be potentially higher utility rates as a result of higher wages to be paid to more qualified individuals.

1.3.5 Workers (relative to ALARA)

No effect on workers relative to ALARA is anticipated.

1.4 Decision

Updated guidance should be provided on the qualification and training of operations personnel.

2. TECHNICAL APPROACH

This section is not applicable to this value/impact statement since the proposed action is an update of previously issued guidance.

3. PROCEDURAL APPROACH

Since the proposed action is an update of information contained in an existing regulatory guide, the only appropriate procedural approach is a revision to the existing guide.

4. STATUTORY CONSIDERATIONS

4.1 NRC Authority

Authority for this guide is derived from the safety requirements of the Atomic Energy Act. In particular, paragraph 50.34(b)(6)(i) of 10 CFR Part 50 requires that applicants for a license to operate a nuclear power plant include information concerning organizational structure, personnel qualifications, and related matters. Section 55.10 of 10 CFR Part 55 requires that operator license applications include information about an individual's education, experience, and related matters.

4.2 Need for NEPA Assessment

The proposed action is not a major action, as defined by paragraphs 51.5(9) and (10) of 10 CFR Part 51, and does not require an environmental impact statement.

5. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICIES

Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure As Low As Is Reasonably Achievable," and Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," reference Regulatory Guide 1.8 with regard to qualification requirements for the Radiation Protection Manager.

There are a number of efforts currently in progress to provide additional reviews and recommendations concerning the subject of personnel qualification that interface with the proposed action (revision of Regulatory Guide 1.8). Efforts in progress within the NRC include the following studies and criteria development tasks.

5.1 Upgrading Qualifications of Operators

The NRC staff provided recommendations in Commission Paper SECY 79-330E, "Qualifications of Reactor Operators," for upgrading the qualifications of licensed operators and senior operators through a program that includes increased training and testing in the areas of thermal-hydraulics and reactor transient response, increased use of simulator training and testing, higher passing grades on licensing examinations, increased experience requirements for operators and senior operators, and increased emphasis on retraining and examination. Based on Commission action on SECY 79-330E, the Office of Nuclear Reactor Regulation by a letter of March 28, 1980, from H. R. Denton to all power reactor applicants and licensees set forth the revised criteria to be used by the staff in evaluating reactor operator training and licensing and established effective dates for their implementation. Licensees and applicants were also informed in the March 28, 1980, letter of other criteria under development that would require additional staff work to produce the necessary requirements that would eventually be established through rulemaking proceedings. Additionally, licensees and applicants were informed that Commission review in the area of operator training and qualification is continuing and may result in identification of additional criteria to be used in evaluation of operators for licensing and relicensing.

5.2 Upgrading Qualifications of Operators

The NRC staff is preparing amendments to 10 CFR Part 55, "Operators' Licenses," and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," that incorporate applicable Commission-approved recommendations of SECY 79-330E and other proposed changes resulting from additional staff review of Part 55 and Part 50. The Regulatory Activities Subcommittee of the Advisory Committee on Reactor Safeguards has reviewed the proposed changes to Part 55 and Part 50. Both the March 28, 1980, H. R. Denton, NRR letter and the proposed rulemaking impact this proposed regulatory guide. With respect to the March 28, 1980, letter, the short-term requirements are either incorporated in the December 6, 1979, Draft Standard ANS 3.1 or noted in the regulatory position of this regulatory guide. With respect to Part 55 and Part 50 revisions, certain positions now included (see Regulatory Positions 1.4a-e) in this regulatory guide will be codified in the regulations in place of the existing general provisions of the regulations. After publication of the final rule changes, Regulatory Guide 1.8 will be revised to eliminate any duplication.

5.3 Onsite and Offsite Managerial and Technical Organizations

The NRC staff has under development criteria for onsite and offsite managerial and technical organizations, including a radiological protection organization, that will provide assurance of the safe operation of a plant during normal and abnormal conditions and the capability necessary to respond to unusual conditions. A contractor was selected (Teknekron Research, Inc.) to assist in the development of criteria listed above. Teknekron Research, Inc., submitted its final report to the NRC in May 1980.

Task I.B.1.1, "Organization and Management of Long-Term Improvements," and Task I.B.1.2., "Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants," of NUREG-0660 describe the criteria development effort for technical and managerial organization. Task I.B.1.2 is a first step in the development of criteria described in Task I.B.1.1. As described in Task I.B.1.2 of NUREG-0660, near-term operating license applicants, as part of the license application review process, are being required to comply with the findings and requirements generated in an NRC interoffice review of licensee organization and management. The review is based, in part, on an NRC document entitled "Draft Criteria for Utility Management and Technical Competence." The first draft of this document was dated February 25, 1980, however, the document continues to change with use and experience in ongoing reviews. The document addresses the organization, resources, training, and qualifications of plant staff and management (both onsite and offsite) for routine operations and accident conditions. The criteria developed for use in evaluation of operator license applicants, as modified by experience from interoffice review of near-term operating license applicants, will form the basis for the final criteria to be developed in accordance with Task I.B.1.1 of NUREG-0660. The criteria reference Regulatory Guide 1.8 and ANS Standard 3.1 with regard to qualifications of nuclear power plant personnel filling the various functional positions described in ANS Standard 3.1. Regulatory Guide 1.8 will be implemented in conjunction with the recommendations of Task I.B.1.1.

5.4 Onsite "Independent Safety Engineering Group"

Each near-term operating license applicant as part of ongoing license application review is being required to establish a group, commonly referred to as the "Independent Safety Engineering Group," that is independent of the plant staff, but is assigned onsite to perform independent reviews of plant operations activities and to evaluate operating experiences at nuclear power plants. Commission Information Paper SECY-80-242 describes the relationship of the proposed "Independent Safety Engineering Group" to other organizational entities such as the Plant Operations Review Committee. The "Draft Criteria for Utility Management and Technical Competence" (see Item 5.3 above) addresses the qualifications of the onsite independent safety engineering group by referencing Section 4.7 of the December 6, 1979, Draft Standard ANS 3.1.

5.5 Shift Technical Advisors

As described in Task I.A.1.1 of NUREG-0660, the NRC is requiring that a technical advisor to the shift supervisor be present on all shifts. Although minimum training requirements have not yet been specified for shift technical advisors presently occupying such positions, shift technical advisors should enhance the accident-assessment function at the plant. By January 1, 1981, shift technical advisors will be required to have a technical education, which is of the quality of courses taught at the college level and includes about 60 semesters hours in basic subjects of engineering and science; specific training in the design, function, arrangement, and operation of plant systems; and training in the expected response of the plant and instruments to normal operation, transients, and accidents, including multiple failures of equipment and operator errors. Commission Information Paper SECY 80-243 addresses the subject of shift technical advisors, including a summary of experience required, duties to be performed, and the number of shift technical advisors required onsite. The qualifications of shift technical advisors as addressed in this regulatory guide are the same as those specified in Commission Paper SECY 80-243. The Commission was also informed in this paper that the Institute of Nuclear Power Operations (INPO) has recently forwarded to NRR for comment a copy of an INPO document, "Nuclear Power Plant Shift Techical Advisor - Recommendations for Position Description, Qualifications, Education and Training" (Enclosure to SECY 80-243), and that the NRC staff had the document under review. Further discussion with INPO may lead to revision of this regulatory guide with regard to shift technical advisors.

5.6 Comparison of NRC, Commercial, and Naval Procedures for Qualification of Personnel

In early 1980, Basic Energy Technology Associates, Inc. (BETA), completed a study for the Office of Nuclear Reactor Regulation that outlined the results of a comparative review of existing NRC requirements, commercial nuclear power plant practices, and the Naval Nuclear Propulsion Program procedures for the selection, training, and qualification of personnel involved in nuclear power plant operation and maintenance. The results of the BETA study entitled "Power Plant Staffing" are documented in NUREG/CR-1280, BETA-103. Public comments were requested on this study. The BETA study, when comparing the majority of practices, used the provisions outlined in ANSI N18.1-1971 and did not consider later versions of the standard. The BETA study will be considered when developing future revisions to this regulatory guide.

5.7 Requirements for Licensing of Operators

The NRC staff has awarded a contract to Analysis and Technology, Inc., for a study of requirements for operator licensing. The scope of the work should result in recommendations for (a) the means to be employed for selection and training of nuclear power plant personnel and the degree to which NRC should be involved in the process, (b) the means to be employed to evaluate the effectiveness of training programs, including who, by job description, should be licensed, (c) the methods to be employed to ensure continued competency of plant personnel and the degree to which the NRC should be involved in establishing the methods, (d) the methods to be employed for maintaining a highly motivated and dedicated work force, and (e) the means to be employed for rapidly requalifying presently licensed operators to meet the proposed new requirements. Task I.A.2.6, "Long-Term Upgrading of Training and Qualifications," of NUREG-0660 provides a description of planned NRC actions related to this contract study. After staff review of the study has been completed, the staff will provide recommendations to the Commission and subsequently factor Commission decisions on the recommendations into a regulatory guide or a regulation.

5.8 Planned NRC Accreditation of Training Institutions

As described in Task I.A.2.7 of NUREG-0660, the NRC staff is conducting a study of procedures and requirements for NRC accreditation of training institutions. A Commission Information Paper on this subject should be completed by late 1980. The NRC staff will also prepare a Commission paper examining various NRC approaches to accreditation of training institutions. Staff action on this later effort should be completed by January 1982.

5.9 Operator Emergency Response and Organization of Personnel

Two other efforts interface with this Regulatory Guide to a lesser extent than the efforts described above. In February 1980, the staff issued for interim use and comment, NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear

Power Plants."* NUREG-0654 was designed to provide a common reference and interim guidance for State and local governments and nuclear facility operators in the development of emergency response plans and preparedness in support of nuclear power plants, as well as to provide guidance to Federal agency personnel. In effect, the thrust of NUREG-0654 is to ensure adequate preparedness planning for protection of the population around nuclear power plants. As a part of this emergency preparedness planning, however, NUREG-0654 necessarily addresses the emergency response capability of the nuclear plant operators. In particular, Section II.B of NUREG-0654 sets forth proposed criteria for the onsite emergency organization of the nuclear plant considered to be necessary to adequately support the plans for offsite response in the event of an accident at the plant.

Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," which endorses, with exceptions, ANSI Standard N18.7/ANS-3.2, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants" (see Regulatory Position 1.1), is being revised to update quality assurance program requirements for the operational phase of nuclear power plants. Regulatory Guide 1.33, while not addressing qualification requirements for plant personnel, addresses organizational requirements.

Both of these efforts, NUREG-0654 and Regulatory Guide 1.33, by addressing staffing and organizational considerations have a potential for impacting Regulatory Guide 1.8.

With regard to the interface of this proposed regulatory guide and the other efforts described above, this version of the regulatory guide is a first step in a comprehensive process of upgrading the training and qualifications of operations personnel. Some of the regulatory guide positions may be eventually displaced by more comprehensive long-term changes (e.g., changes to NRC regulations) in the area of personnel qualification; however, these positions are expected to be consistent with the long-term changes.

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*NUREG-0654 (FEMA-REP-1) is available at current rates through the Government Printing Office Sales Program, Distribution Services Section, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 or from the National Technical Information Service, Springfield, Virginia 22161.

6. SUMMARY AND CONCLUSIONS

A draft revision to Regulatory Guide 1.8, "Selection and Training of Personnel," should be prepared. The draft revision should endorse, with certain exceptions and supplements, the December 6, 1979, Draft Standard ANS 3.1. (A discussion of proposed regulatory positions in second proposed Revision 2 to Regulatory Guide 1.8 is provided in the attachment to this value/impact statement.)

RATIONALE FOR REGULATORY POSITIONS

1. Regulatory Position 1.1 addresses the applicability of documents referenced in Draft Standard ANS 3.1. Regulatory Position 1.1 also provides information on a regulatory guide being prepared on nuclear power plant simulators for use in operator training.

This position has been provided, as is customary in many regulatory guides, to assist the user of the guide in determining the applicability of referenced documents.

2. Regulatory Position 1.2.1 takes exception to Section 3.1 of the draft version of ANS 3.1 concerning personnel temporarily filling positions owing to absences of the principal. The standard permits the use of temporary personnel replacements for a period not to exceed 3 months. The regulatory position states that temporary personnel should not be used as replacements for periods exceeding 1 month.

The 3-month period allowed by the standard is too long to be considered temporary. If the principal is replaced for a period greater than 1 month, the replacement should not be considered temporary and qualifications of the replacement should be subject to the provisions of Section 4 of the standard.

Since temporary absences normally do not exceed one month, the position should not result in impact on licensees. The provision exists in the standard to recognize that a principal may be temporarily absent. The regulatory position should result in maintaining a consistent level of staff qualifications.

3. Regulatory Position 1.2.2 is intended to provide practical guidance on how the standard should be applied to contractor personnel. The proposed regulatory position does not take exception to ANS 3.1. While Section 1, "Scope," of the standard states that the standard is limited to personnel within the owner organization, the definition of owner organization states that contracted personnel are included in the owner organization. Thus contracted personnel are within the scope of Draft Standard ANS 3.1.

4. Regulatory Position 1.3 defines the term "college-level education" that is used throughout the standard. It states that "college-level education" is construed to mean course work satisfactorily completed (i.e., finished with a grade at the 70% level) at or conducted by a college or university with curricula accredited by a nationally recognized agency such as the Accreditation Board for Engineering and Technology. The regulatory position is provided to establish a norm for the provisions of ANS 3.1 dealing with "college-level education" requirements and is intended to describe the quality of the education to be completed.

Case-by-case reviews of educational programs conducted at nonaccredited training institutions pending additional guidance on accreditation will require additional regulatory staff review. The regulatory position indicates that the NRC intends to establish alternative means of accreditation of training programs and that course work completed under such programs may become acceptable.

College-level education is specified in terms of semester hours of college-level subjects. To ensure that there is no misunderstanding as to the meaning of a college credit hour, the following example is repeated from Dr. Miller's letter of May 21, 1980, to Mr. H. Denton on this subject. Dr. Miller, Chairman, Nuclear Engineering Program of Ohio State University, states: "There appears to be some misunderstanding as to the meaning of a college credit hour. Specifically the 60 semester hours of college level subjects specified for senior operators in fact implies approximately 900 classroom hours. However, college instructors normally assume they will be assigning approximately 2 hours of work outside the classroom for every hour in the classroom. As a consequence, 60 semester hours implies approximately 2700 hours of work for the average student. Obviously, students of superior ability will complete them in fewer hours and vice versa." The NRC staff agrees with Dr. Miller's comments concerning the meaning of a college credit hour. Therefore, 60 semester hours (representing approximately 2700 hours of work for the average student) was selected as the established norm for judging education requirements.

5. Regulatory Position 1.4 lists exceptions taken to ANS 3.1 to provide for consistency with the anticipated amendments to 10 CFR Part 55 and 10 CFR

Part 50, which are presently undergoing NRC review prior to publication for public comment.

6. Regulatory Position 2.1 provides guidance on use of the provisions of Section 4.1 of ANS 3.1 for conducting case-by-case evaluation of an individual's qualifications when the individual does not meet those qualifications stated in the standard. Although it was recognized in the development and review of the draft version of ANS 3.1 that there may exist situations where individuals could become qualified for a position based on factors other than those included in the standard (i.e., education, experience), this position recognizes this situation should be the exception rather than the rule. This position will result in fewer deviations from the requirements of the standard. Additionally, this position helps ensure that such deviations are not used for a number of specific plant positions that are responsible for overall plant safety.

This regulatory position, if not initially applied with caution, could have a significant impact on licensees. A phased implementation of the standard and the regulatory guide is necessary so that a high turnover rate of personnel does not ensue that might adversely affect the safe operation of a plant. The position most significantly affected by upgraded qualifications is that of shift supervisor. Appendix A to this regulatory guide provides a detailed discussion of shift supervisor qualifications including recommendations for implementation.

As indicated in the implementation section of the regulatory guide, the regulatory guide will be applied in conjunction with Task I.B.1.1, "Organization and Management of Long-Term Improvements," of NUREG-0660. Task I.B.1.1 provides for licensee submission by May 1981 of a plan for implementing the NRC requirements for upgrading of management and technical resources whether or not an effective active guide on personnel qualification has been issued by that date. Thus, the impact on each licensee will be determined on a case-by-case basis.

Once fully implemented, this regulatory position should provide assurance of a consistent level of staff qualifications.

7. Regulatory Position 2.2.1 provides limitations concerning acceptability of certification of plant managers and technical managers at the plant or at an appropriate simulator as a condition of qualification. It states that this certification should either be conducted or approved by the NRC staff. This

regulatory position provides for regulatory staff involvement in the certification process since the provision of the standard for such certification is intended as an alternative to holding a senior operator license. The position should not significantly impact licensees; however, NRC staff time will be required in conducting or approving certification examinations.

8. Regulatory Position 2.2.2 states that the recommendation of the standard for the maintenance manager to have nondestructive testing familiarity, craft knowledge, and an understanding of electrical, pressure vessel, and piping codes and standards should be construed as a requirement.

These qualifications are essential to the performance of the duties of a maintenance manager. The most significant impact of this position may be that the provision of the standard is very general and application of the general requirement may not be applied uniformly. This could result in additional staff review time.

9. Regulatory Position 2.2.3.1 presents the minimum formal training a radiation protection manager should receive while Section 4.4.4 of ANS 3.1 states only that a manager must receive formal training. This position further clarifies the intent of a similar position contained in the first proposed revision of Regulatory Guide 1.8 issued for comment in February 1979.

10. Regulatory Position 2.2.3.2 specifies that a certified (ABHP) health physicist's experience be in a position of supervisory capacity in a health physics program for an operating nuclear power plant because the NRC staff does not believe that a health physicist who has no such experience should qualify as a radiation protection manager.

11. Regulatory Position 2.2.3.3 has been added to provide clarification that the 4 years of experience required by the standard for the Radiation Protection Manager should be above the technician level and that the 3 years of experience at a nuclear facility required by the standard should involve directing the activities of radiation protection technicians. The provision for 3 years of experience directing the activities of radiation protection technicians is consistent with the first proposed revision of Regulatory Guide 1.8 issued for comment in February 1979. The provision that 4 years of experience be above

the technician level when coupled with the provision for 3 years of experience directing the activities of radiation protection technicians represents only a minor modification of present staff practice and is consistent with the requirement for 4 years of professional-level experience for other group leaders.

12. Regulatory Position 2.3.1 addresses qualifications of shift supervisors. See Appendix A to the guide for a detailed discussion of this regulatory position.

13. Regulatory Position 2.3.2 provides for corporate management involvement in the certification of the competency of shift supervisor candidates and for documentation of such certification. This provision is similar to the provisions for certification by corporate management of the competency of operators and senior operators.

To ensure proper conduct of training programs and qualification of plant personnel, corporate management of each licensee should establish a definitive presence and involvement in the selection, training, and qualification of operations personnel. Such a provision should have minimal impact on licensees. This particular position is responsive, in part, to one of the subitems of Task I.B.1.1, "Organization and Management of Long-term Improvements," of NUREG-0660.

14. Regulatory Position 2.4.1 makes experience requirements for onsite professional-technical group leaders more consistent than past guidance. The regulatory position is self-explanatory. The overall experience requirements for technical group leaders should be more consistent.

15. Regulatory Position 2.4.2 is self-explanatory. The standard contains a potential loophole that could allow the position of chemistry and radiochemistry group leader to be filled without the person having had nuclear power plant experience. In order to ensure that this situation will not be permitted, the equivalence of the training program to experience should be allowed only to a maximum of 6 months.

16. Regulatory Position 2.5 states that training for the Training Coordinator should include the training required by Section 5.4, "General Employee Training,"

of ANS 3.1. This type of training is required by ANS 3.1 for other professional-technical group leaders and such training is appropriate for the Training Coordinator.

17. Regulatory Position 2.6 takes exception to the requirement of Section 4.4.7.2 of ANS 3.1 that "Instructors who provide instruction on the simulator shall hold a senior operator license for a similar unit (PWR, BWR, HTGR) or have been certified at an appropriate plant simulator...." The proposed regulatory position is consistent with the staff recommendation approved by the Commission that training center and facility instructors who teach systems, integrated responses, transient, and simulator courses should demonstrate their competence to NRC by successful completion of a senior operator examination and should participate in appropriate requalification programs to ensure they are cognizant of current operating history, problems, and changes to procedures and administrative limitations (see Commission Paper SECY-79-330E).

This provision was contained in the March 28, 1980, letter from H. Denton to all power reactor applicants and licensees.

18. Regulatory Position 2.7 notes that the requirements of Section 4.4.8 of Draft Standard ANS 3.1 concerning Shift Technical Advisor qualifications are not consistent with the requirements for the Shift Technical Advisor previously forwarded to all applicants and licensees. (Dates of letters to licensees and applicants are indicated in the proposed regulatory position.) The proposed regulatory position states that the previously issued requirements are included as Appendix B to the guide and should be followed. The proposed regulatory position provides for consistency with present staff practice.

19. Regulatory Position 2.8 is an amplification of the requirements contained in Sections 5.3.4 and 5.4 of the standard for training programs for radiation protection technicians. The guide position should be helpful in establishing the content of the training program for radiation protection technicians. The guide position does not represent a significant change from the more general provisions of Draft Standard ANS 3.1.

20. Regulatory Position 2.9 states that training in accordance with Section 5.4, "General Employee Training," should be provided for the "Engineer-in-Charge"

independent of whether this is an onsite or offsite position. General employee training is provided to all persons regularly employed in a nuclear power plant. Such training is essential for the "Engineer-in-Charge" since this person is to provide additional services to the plant beyond those provided by the operations organization professional-technical personnel. It cannot be envisaged that such a person can successfully perform his or her required duties without training in the subjects covered in Section 5.4 of Draft Standard ANS 3.1.

21. Regulatory Position 3.1 states that a position task analysis should be performed for all operations personnel in which the tasks performed by the person in each position are defined, and the required training, in conjunction with education and experience, is identified to provide assurance that the tasks can be effectively performed. For many positions described in Draft Standard ANS 3.1, a position task analysis is a requirement of the standard.

This proposed regulatory position is consistent with Task I.A.2.2, "Training and Qualifications of Operations Personnel," of NUREG-0660. NUREG-0660 states that the Office of Nuclear Reactor Regulation will issue a requirement by October 1, 1980, for each licensee to review its training program for all operations personnel, including maintenance and technical personnel and to justify the acceptability of training programs on the basis that these programs provide sufficient assurance that safety-related functions will be effectively carried out. NUREG-0660 states that the preferred method of fulfilling this provision is a position task analysis.

In the February 1980 report to the AIF Policy Committee on Follow-up to the Three Mile Island Accident by the Working Group on Action Plan Priorities and Resources, it was indicated that tasks analyses of positions were worthwhile and should be pursued jointly by the NRC and INPO, with INPO performing task analyses for those positions that are generally used industry wide. The working group also provided the following additional comments:

> "These analyses would be conducted by professionals and include recommendations for qualifications and training needed for a particular position. Each utility would then evaluate in a similar manner any unique position in their organization not addressed in the INPO study. These

studies by INPO should be completed by early 1981 and would greatly reduce individual utility costs.

"Evaluating positions affecting safe operation other than the licensed operators to ascertain that their training and qualifications are adequate has considerable merit. It would introduce no plant safety hazards and would have a small degree of positive effect on reducing overall risks if a need for additional training is identified. The cost for a utility-unique analysis would be an unnecessary use of resources for each utility at an aggregate cost which is large."

The NRC staff agreed with the AIF working group comments and revised NUREG-0660 to state that the task is amenable to a generic approach in which INPO could perform task analyses of those positions generally used throughout industry and each utility could evaluate in a similar manner any unique position in its organization not covered by the INPO study.

22. Regulatory Position 3.2.1 incorporates Task II.B.4, "Training for Mitigating Core Damage," of NUREG-0660 as a proposed regulatory position. The Commission on February 7, 1980, approved this item in conjunction with other "Near-Term Operating License Requirements" (see Table A.1 of NUREG-0660) as being necessary to implement before full power operation is permitted for near-term operating license applicants. Task II.B.4 of NUREG-0660 states NRR will issue a requirement by October 1, 1980, for licensees to develop a training program by January 1, 1981, and implement the training program by April 1, 1981.

The AIF Working Group on Action Plan Priorities and Resources stated that the inclusion of this item in the periodic retraining program will increase its potential for effectiveness.

23. Regulatory Position 3.2.2 incorporates Task I.A.2.5, "Plant Drills," of NUREG-0660 as a proposed regulatory position. Task I.A.2.5 indicates that requirements for conduct of plant drills will be issued by January 1981 and that drills will begin at operating reactors by July 1, 1981. Operating

licensee applicants will begin drills by July 1, 1981, or before operating license issuance, whichever is later.

The AIF Working Group on Action Plan Priorities and Resources indicated plant walk-throughs and drills would be effective in training.

24. Regulatory Position 3.2.3 takes exception to Section 5.5.1.3.1 of Draft Standard ANS 3.1 concerning criteria for requiring a licensed individual to participate in accelerated requalification. This proposed regulatory position was included to provide consistency with instructions that were included in the NRR H. R. Denton letter of March 28, 1980, to all reactor applicants and licensees.

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REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

REGULATORY GUIDE 5.60

STANDARD FORMAT AND CONTENT OF A LICENSEE PHYSICAL PROTECTION PLAN FOR STRATEGIC SPECIAL NUCLEAR MATERIAL IN TRANSIT

USNRC REGULATORY GUIDES

Regulatory Guides are issued to describe and make available to the public methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

Comments and suggestions for improvements in these guides are encouraged at all times, and guides will be revised, as appropriate, to accommodate comments and to reflect new information or experience. This guide was revised as a result of substantive comments received from the public and additional staff review.

Comments should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch.

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INTRODUCTION

This regulatory guide describes the standard format and content suggested by the Nuclear Regulatory Commission (NRC) for use in preparing Physical Protection Plans for strategic special nuclear material (SSNM) in transit. By using this guide for preparing a Physical Protection Plan and Physical Protection Arrangements for Specific Shipments, the licensee will minimize administrative problems associated with the submittal, review, and approval of the plan. Conformance with this guide is not required by the NRC. A licensee who uses a format that will provide an equal level of completeness and detail and that will satisfy the requirements of the applicable NRC regulations may use his own format. However, the format and content herein presented are acceptable to the NRC staff.

The Atomic Energy Act of 1954, as amended, directed the Atomic Energy Commission (AEC) to regulate the receipt, manufacture, production, transfer, possession, use, import, and export of special nuclear material (SNM) in order to protect the public health and safety and to provide for the common defense and security. The Energy Reorganization Act of 1974 transferred all the licensing and related regulatory functions of the AEC to the NRC.

The principal requirements for physical protection of licensed activities against theft and radiological sabotage of SSNM in transit are contained in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Part 70, "Domestic Licensing of Special Nuclear Material," and Part 73, "Physical Protection of Plants and Materials."

Paragraph 50.34(c) of 10 CFR Part 50 and paragraphs 70.22(g) and 70.22(h) of 10 CFR Part 70 identify the physical protection information that must be provided in a Physical Protection Plan as part of a license application. This plan is required in order for the licensee to comply with specific physical protection requirements of 10 CFR Part 73 and must be submitted with each application for a licensee to possess or process SSNM or for a license authorizing the transport or delivery of SSNM.

The information in this guide applies to any NRC licensee or applicant for an NRC license who anticipates transporting or delivering uranium-235 (contained in uranium enriched to 20 percent or more in the U-235 isotope), uranium-233, or plutonium alone or in any combination in a quantity of 5000 grams or more, computed by the formula: grams = (grams contained U-235) + 2.5 (grams U-233 + grams plutonium).

As developments and changes in the nuclear industry occur, the Commission's requirements for information may need modification. Revisions to this Standard Format will be made, as necessary, to accommodate these changes.

Purpose and Applicability

This regulatory guide has been prepared as an aid to uniformity and completeness in preparing and reviewing the Physical Protection Plan, which is a prerequisite for a license for transporting or delivering SSNM. Amplification and clarification of the protection arrangements to be made for each individual shipment of SSNM are also provided.

The information requested in this Standard Format is the minimum necessary for an acceptable Physical Protection Plan. Additional information may be required during the staff review of a particular plan. It is also the licensee's responsibility to be aware of applicable new and revised NRC regulations. The information provided should be up to date with respect to the state of technology for the physical protection techniques and systems that the licensee proposes to use.

Information and procedures that are delineated in the regulatory guides in Division 5, "Materials and Plant Protection," and that are appropriately contained in certain sections of the licensee's Physical Protection Plan for fixed sites may be incorporated by reference.

Upon receipt of a Physical Protection Plan for review and approval, the NRC staff will perform a preliminary review to determine whether the plan is reasonably complete. The staff will use the Standard Format as a guideline to identify the necessary information. If a licensee does not provide a reasonably complete presentation of the necessary information, further review of the plan may be suspended until the needed information is provided.

Organization of This Document

This document is divided into two parts. Part I provides the standard format and content to be used for submittals of the Physical Protection Plans for SSNM in transit. Each section in Part I is referenced to the portions of the regulations that apply to that section, the reference provided in brackets following the section heading. Also, the sections in this part are divided into two subsections entitled "Intent" and "Content." The "Intent" subsections are exclusively explanatory in nature and are provided to give the licensee additional information regarding the purpose for including such provisions in the regulations and the relationships between the different provisions. Examples of subsystems and procedures that may be used in different situations to achieve the capabilities required in the regulations are also provided, including references to pertinent sections of the reference safeguards system described in §73.26. This type of information should aid the licensee in designing a well-balanced, integrated physical protection system that can meet all the performance capability requirements. The "Content" subsections provide detailed guidance on the information to be included in the Physical Protection Plan submittals and the standard format to be used to ensure that the plans submitted are complete and in a form that can be easily reviewed by the NRC staff.

Part II contains standard format and content information relating to submittals of Physical Protection Arrangements for Specific Shipments. These submittals are intended to supplement the Physical Protection Plan submittal with changes in physical protection subsystems and procedures that will be effective for a given shipment plus specific scheduling information and other arrangements that cannot be determined very far in advance of the date for a specific shipment.

In addition to the Physical Protection Plan and the arrangements for specific shipments, licensees are also required to submit plans for guard qualification and training programs and Safeguards Contingency Plans. All these plans together comprise the physical protection system required to be provided by licensees who are involved in the transportation of strategic special nuclear material; therefore, they all should be consistent with one another.

Use of the Standard Format

The licensee should follow, as nearly as possible, the numbering system of the Standard Format in Parts I and II of this guide. If certain subsections are not applicable, this should be clearly stated and sufficient information should be provided to support that conclusion. The licensee may expand on the information provided, as necessary, to adequately cover any safeguards measures unique to the licensee's operation.

Upon completing the plan, the licensee should use the table of contents of the Standard Format as a checklist to ensure that each subject has been addressed.

Style and Composition

A table of contents should be included in each submittal.

The licensee should clearly and concisely present all information. Confusing or ambiguous statements and general statements of intent should be avoided. Definitions and abbreviations should be consistent throughout the submittal and in keeping with generally accepted usage.

Wherever possible, duplication of information should be avoided. Thus, information already included in other NRC-required physical protection plans, e.g., the licensee's Site Physical Protection Plan and Safeguards Contingency Plan, may be covered by specific reference to the appropriate sections of those documents.

Drawings, diagrams, maps, and tables should be used when information may be presented more adequately or conveniently by such means. These illustrations should be located in the section where they are first referenced or in appropriately referenced appendices. All information presented in drawings should be legible, symbols should be defined, and drawings should be large enough (not unduly reduced) to be read by an individual having normal vision.

Physical Specifications of Submittals

All material submitted in an application should conform to the following physical dimensions of page size, quality of paper and inks, numbering of pages, etc.:

a. Paper Size

Text pages: 8-1/2 x 11 inches.

Drawings and graphics: 8-1/2 x 11 inches is preferred; however, a larger size is acceptable if the finished copy when folded does not exceed 8-1/2 x 11 inches.

b. Paper Stock and Ink

There should be suitable quality in substance, paper color, and ink density for handling and for reproduction by microfilming.

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c. Page Margins

A margin of no less than 1 inch should be maintained on the top, bottom, and binding side of all pages submitted.

d. Printing

Composition: text pages should be single spaced.

Type face and style: must be suitable for microfilming.

Reproduction: may be mechanically or photographically reproduced. All pages of the text may be printed on both sides. Images should be printed head to head.

e. Binding

Pages should be punched for looseleaf ring binding.

f. Page Numbering

Pages should be numbered sequentially within each section. Do not number the entire report sequentially.

g. Format References

References to this Standard Format should be made by part, chapter, and section number.

Procedures for Updating or Revising Pages

The updating or revising of data and text should be on a replacement-page basis.

The changed or revised portions of each page should be highlighted by a vertical line. The line should be in the margin opposite the binding margin for each line changed or added. All pages submitted to update, revise, or add pages to the report should indicate the date of the change. The transmittal letter should include an index page listing the pages to be inserted and the pages to be removed. When major changes or additions are made, pages for a revised table of contents should be provided.

Number of Copies, Address, and Schedule for Submittal

The licensee should submit six copies of his Physical Protection Plan to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, in accordance with the provisions of paragraph 73.20(c) of 10 CFR Part 73.

Public Disclosure and Classification

The NRC has determined that the public disclosure of the details of physical protection plans and programs is not in the public interest. Such details are withheld in accordance with paragraph 2.790(d) of 10 CFR Part 2.

Thus the Physical Protection Plan for transport of SSNM and individual submittals of Physical Protection Arrangements for Specific Shipments should be appropriately marked and handled accordingly. Other related proprietary or classified information should be clearly identified and submitted in accordance with applicable directives. Each such submittal of proprietary or classified information should be accompanied by the licensee's detailed reasons and justifications for requesting exemption from public disclosure, as required in paragraph 2.790(b) of 10 CFR Part 2.

Compatibility

The licensee should ensure that the Physical Protection Plan for transport of SSNM is compatible with applicable parts of all other physical protection plans, including the Site Physical Protection Plan, separate submittals of Physical Protection Arrangements for Specific Shipments, the Safeguards Contingency Plan, and plans for guard qualification and training programs.

PART I

PHYSICAL PROTECTION PLAN

Chapter 1 GENERAL ISSUES

1.1 Purpose and Scope

Intent

The primary purpose of a licensee's plan for the physical protection of strategic special nuclear material (SSNM) in transit is to assist in the NRC review of the adequacy of the licensee's physical protection system relative to the particular routes or itineraries proposed in the plan. However, after the plan has been approved by the NRC, it could become a guide for the licensee's security personnel in the implementation of the licensee's physical protection system for SSNM in transit.

Content

Describe the purpose of the Physical Protection Plan and characterize its contents. Indicate the additional types of information to be found in the Physical Protection Arrangements for Specific Shipments (prepared for each shipment in accordance with Part II of this guide) that would complete the description of the physical protection system for a given shipment and that would describe in detail the actual shipment itinerary and schedule.

1.2 Safeguards Contingency Plan
[§73.25(d)(1)(ii), (d)(1)(iii), Appendix C to Part 73]

Intent

A predetermined plan to respond to safeguards contingency events is required to be prepared, based on personnel and other physical protection resources described in the Physical Protection Plan for SSNM in transit. Specific requirements for the contingency plan are provided in Appendix C, "Licensee Safeguards Contingency Plans," to 10 CFR Part 73. Regulatory Guide 5.56, "Standard Format and Content of Safeguards Contingency Plans for Transportation," provides guidance for the preparation of transportation contingency plans. The licensee is reminded that all three submissions--the Physical Protection Plan, the Physical Protection Arrangements for Specific Shipments, and the Safeguards Contingency Plan--together describe the system for physical protection of each particular shipment. They should be developed and maintained to be completely consistent with each other for each shipment.

Content

Describe briefly the Safeguards Contingency Plan prepared in accordance with Appendix C to 10 CFR Part 73. Relate it to the Physical Protection Plan and Physical Protection Arrangements for Specific Shipments.

1.3 Security Organization Training, Equipment, and Qualifications
[§73.25(d)(1)(i), (d)(1(iv), (d)(1)(v), Appendix B to Part 73]

Intent

The purpose of these requirements for the qualification and training of security organization personnel is to ensure that security organization members will be fully qualified, equipped, and trained and thoroughly familiar with the procedures and equipment they will use during the progress of a shipment. These requirements are detailed in Appendix B, "General Criteria for Security Personnel," to 10 CFR Part 73 and are cited also in the reference system [§73.26(d)(4) and (d)(5)]. The requirements generally are that each security organization member be able to perform at least those duties described in the Physical Protection Plan that he might reasonably be assigned.

The specific description of qualification and training of security organization members will be provided as part of a separately prepared and submitted document entitled "Transportation Security Organization Training, Equipment, and Qualifications Plan." That plan should be consistent with the Physical Protection Plan with regard to the numbers of personnel, the equipment described, and the tasks to be performed by security organization personnel. In particular, the physical and performance characteristics of firearms carried by or immediately available to armed escorts and armed response personnel are described in Appendix B to 10 CFR Part 73. Although all three types of weapons described--semiautomatic rifles, shotguns, and sidearms--would normally be expected to be carried by, or immediately available to, these personnel for the physical protection of road and rail shipments and at transfer points, sidearms alone would normally be considered sufficient armament for armed escorts on aircraft and ships.

Content

Affirm that a Transportation Security Organization Training, Equipment, and Qualifications Plan has been submitted to the NRC according to the requirements set forth in Appendix B to 10 CFR Part 73. Affirm also that such plan is consistent with this plan as to equipment and the number of security personnel required to implement the physical protection system described herein. Briefly relate the two plans to ensure that all members of the security organization will be adequately qualified and trained to perform the duties normally assigned them. Refer to the Transportation Security Organization Training, Equipment, and Qualifications Plan to describe the equipment (e.g., armored escort vehicles, firearms, night vision equipment) that will be used by the escorts to protect themselves as well as to perform the tasks assigned to them.

1.4 Security Management
[§73.25(d)(1)(i)]

Intent

Security organization management is a crucial component of the physical protection system. Some specific security management considerations are provided in paragraphs 73.26(d)(2) and (d)(3) of the reference system. The management of the security organization is the vital link between the licensee and those responsible for direct implementation of the overall plan for physical protection of a shipment. Thus, it is necessary to have a detailed description of the licensee's plan for management of the security organization to understand fully how the Physical Protection Plan will be implemented.

Content

Refer to the Security Management section of the Transportation Security Organization Training, Equipment, and Qualifications Plan to briefly describe the security organization management structure. Affirm that this structure will ensure that the physical protection system will not fail to function for lack of the presence of a member of the security organization with proper authority to conduct security-related functions performed at the movement control center.

1.5 Testing and Maintenance Program
[§73.20(b)(3)]

Intent

This provision requires a testing and maintenance program. This program is to ensure that all activities and devices on which the physical protection system depends to maintain shipment security continue to be available to the physical protection system in a reliable and effective manner. It also is intended to ensure that the level of protection represented in the Physical Protection Plan submitted does not deteriorate through neglect, disuse, or inappropriate substitution of system components.

Content

Describe the program to regularly test and inspect all components (design features, equipment, vehicles, and procedures) of the physical protection system to ensure that the design, construction, and operation of all security-related components will continue to be substantially as described in the currently approved Physical Protection Plans. Include a description of how these programs will be managed, the lines of authority for such management, and the schedule for licensee management review of the adequacy of the physical protection system.

Describe the program to provide routine preventive maintenance for all security-related equipment, vehicles, and design features in the physical protection system. Include the components covered by the program and the operations and supervisory personnel who will implement the program. Also describe the capability of the escort force to make emergency repairs en route. Describe the circumstances under which such emergency repairs might be required.

1.6 Security Records
[§73.70]

Intent

Certain types of records regarding shipments of SSNM must be kept (see §73.70). These requirements are intended to produce a verifiable record to show that appropriate detailed planning functions were carried out for each shipment and to expedite timely and efficient tracing of any missing shipment.

Content

Describe the recordkeeping program designed to keep track of all licensee personnel who were authorized access to SSNM in transit, other persons who were authorized access, other persons who may have had access to SSNM in transit, and other security-related matters. Include documentation of security assignments, tests, inspections, and audits performed on the physical protection system. Also include logs of previous shipments and other information relating to specific routes and modes of transport such as carriers, major roads used, or flight numbers, as specifically required by §73.70.

1.7 Reports to NRC [§73.71]

Intent

This provision is designed to ensure that the NRC will be apprised of all security-related incidents that occur during shipments of SSNM. Thus the NRC will be able to continually update its knowledge, both of threats posed to such shipments and of the ability of existing physical protection systems to deal with these threats.

Content

Describe the procedures to ensure that NRC is provided with a full accounting of all contingency incidents necessitating reports to the NRC, as detailed in §73.71. Include a description of the management structure and procedures to ensure compliance.

1.8 Redundancy and Diversity [§73.20(b)(2),§73.25(d)(4)]

Intent

It is required that the physical protection system be designed with redundant and diverse measures. This is to ensure that the system will remain capable of providing the necessary level of protection under adverse conditions that could cause the failure of some system elements. Redundancy means providing several measures (which may be the same measure duplicated) to perform the same function or subfunction. This would prevent failure of the entire system if there were failure of one or more system elements.

Diversity means providing several different kinds or types of measures that contribute to the performance of a particular security function or subfunction. Providing different measures with differing characteristics (e.g., sensitivities, failure modes, strengths, weaknesses) ensures continued performance of a given capability despite adverse operational conditions, attempts at system sabotage, or exploitation of a given measure.

The reference system provides redundancy in communications between the escort force and the movement control center (paragraph 73.26(i)(6) for road shipments and paragraph 73.26(k)(4) for rail shipments). The reference system also suggests "corrective action procedures and compensatory measures" to ensure that "the effectiveness of the physical protection system is not reduced by any single failure or other contingencies affecting the operation of the physical protection related equipment or structures" [paragraph 73.26(h)(5)].

The licensee also must be able to communicate with local law enforcement authorities if the escorts are attacked. Routine operating procedures must be designed to ensure that a single adversary attack cannot destroy the armed escorts' capability to notify the local law enforcement authorities of the need for assistance [paragraph 73.25(d)(4)]. (This would be accomplished in the case of road shipments, for example, by proper spacing and randomly varying the distances between the transport and the escort vehicles.) The assurance of the survivability of a communications capability would normally be included under the general redundancy and diversity requirements of paragraph 73.20(b)(2). However, the integrity of this particular communications capability is considered so crucial to the primary mission of the physical protection system that it has been stated as a separate capability in paragraph 73.25(d)(4). This capability is also referred to in the reference system [paragraph 73.26(f)(2)] in the context of transfer operations procedures. Generally, the licensee has the responsibility for determining where measures for redundancy and diversity may need to be designed into the physical protection system and for designing the system accordingly.

Content

Describe the portions of the physical protection system that have been reinforced with redundant or diverse measures to provide assurance of continued capability to protect a shipment during adverse conditions or attacks. Include, in particular, a description of physical protection subsystems and procedures that ensure that a single adversary attack cannot keep the armed escorts from notifying local law enforcement authorities of the need for assistance.

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Chapter 2 SHIPMENT PLANNING AND CONTROL

2.1 Preplanning of Shipment Itineraries [§§73.25(b)(1)(i) and (d)(3)]

Intent

The intent of this requirement for preplanning of shipment itineraries is to decrease the vulnerability of SSNM shipments and to increase the ability of response forces to assist the escorts if a problem arises. Route preplanning can help accomplish this objective. It can determine areas where a shipment would be vulnerable to attack (e.g., scenes of recent natural disasters, labor dispute areas) and avoid these areas. It can minimize the stops required. It can minimize layover (storage) time. It can select routes allowing minimum interruption in communications with the movement control center. The route itinerary and scheduling information resulting from preplanning can familiarize escort and movement control center personnel with the route for a particular shipment. This familiarity will help ensure quick response to any emergencies that may arise.

Establishing liaison with local law enforcement authorities (LLEAs) to arrange for assistance along routes will ensure that the LLEAs are familiarized beforehand with the types of assistance they could be asked to provide SSNM carriers.

The preplanning function will usually involve the determination and detailed consideration of several alternative routes and rest stops along each route that generally would be acceptable for SSNM shipments. However, the final selection of one of these alternative routes or of particular rest stops for a specific shipment will depend on the most current information available about conditions along each route. Such information will be obtained and maintained according to the procedures described in Section 2.2 of the Physical Protection Plan (see Section 2.2 of Part I of this guide). Sufficient flexibility should be incorporated in the planning of shipments to avoid regular patterns as suggested in the reference system [paragraph 73.26(b)(1)].

Other specific considerations related to the preplanning capability are in the following sections of the reference system: paragraphs 73.26(b)(2) and (3) and 73.26(c) for general information; paragraph 73.26(i)(1) for road shipments; paragraphs 73.26(j)(1), (2), (4), and (8) for air shipments; paragraphs 73.26(k)(1) and (3) for rail shipments; and paragraphs 73.26(l)(1), (2), (3), (5), (6), and (8) for shipments by sea.

Content

1. Planning and Scheduling Considerations. Discuss provisions to ensure that the transit times of SSNM shipments will be minimized and that routes will be selected to avoid areas of natural disaster, civil disorder, or other similar threats to the security of the shipment. Also discuss provisions to ensure that shipments will encounter minimal, if any, storage times and that delivery will be made to the receiver as soon as possible after arrival at the final destination.

2. Itinerary Information. The itineraries (or routes) described in this section of the plan should show the following detail: (a) that the licensee has a comprehensive understanding of conditions and situations that may occur on the route, (b) that the licensee will use this knowledge to avoid areas imposing excessive and unnecessary demands on the escort force or unnecessarily increasing the vulnerability of the shipment, and (c) that the licensee will be prepared to take any necessary action when the shipment must pass through areas of high vulnerability.

Information on planned itineraries routinely used in the transport of the SSNM covered by this Physical Protection Plan should be presented in this section, either directly or by referring to the pertinent appendices of the plan. For brevity and clarity of presentation in the body of the plan, this information is best presented in appendices. Information contained in the licensee's Safeguards Contingency Plan may be referenced to the extent it is applicable. Separate appendices or sections should be used for each itinerary. These may be referred to in preparing submittals of Physical Protection Arrangements for Specific Shipments while noting any changes or updating route information, as necessary. (See Regulatory Guide 5.57, "Shipping and Receiving Control of Special Nuclear Material." See also Regulatory Guide 5.56 for examples of charts and tables describing itineraries.)

For each route, the itinerary information should include route segment identifications; pertinent time and distance factors; and locations of LLEA facilities, possible rest stops, transfer points, and points where there are changes in transportation mode. If some of the information requested is not available at the time the plan is submitted, it may be deferred and included instead in the Physical Protection Arrangements for Specific Shipments. Tables and charts needed for presentation of the itinerary information are described below:

a. Route Overview--a graphic display of the overall route, including road, rail, air, and sea transportation segments. It may be presented on one or more sheets. It may be a line diagram or it may be depicted on a road or topographical map. The map or diagram should be large enough in scale to indicate clearly the pertinent information just described. The planned route segments should be indicated clearly on the display. They should be easily correlated with the pertinent LLEAs and recognized geographic features (e.g., roads, terminals, waterways) forming the boundaries of the segments or otherwise referred to in the plan. The segments may be defined according to the boundaries of the jurisdictions of the pertinent LLEAs. The origin and destination of the itinerary should be clearly indicated. Alternative route segments may be appropriately indicated as alternative route segments of the original route or may be described separately. Route segments should be numbered, or otherwise identified, by an appropriate code.

b. Route Segment Descriptions--a table showing the following information for each route segment:

(1) Number or identification code of the route segment.

(2) Segment boundaries (start and end points).

(3) Number of persons included in escort force. Describe also the source and number of armed response personnel other than escort force members expected to be available at transfer or other points included in the segment. Refer, if necessary, to the Transportation Security Organization Training, Equipment, and Qualifications Plan.

(4) Average estimated time to complete the segment.

(5) Provisions for escort force personnel to sleep or rest during traversal of the segment. Locations of possible refueling and rest stops on the route.

(6) Departure points for leaving planned route and embarking upon alternative route segments.

c. Liaisons with LLEAs--a table listing the LLEA information for each route segment. Include the LLEA organization name and the person contacted (plus his title and phone number within the organization). Briefly describe the arrangement that exists between the LLEA and the licensee concerning possible calls for assistance. Index this information to the numbering or other code used for identifying route segments above.

The licensee should describe the information given LLEA organizations to familiarize their personnel with the unique types of challenges they may face if called to aid a shipment carrier within their jurisdiction.

2.2 Maintaining Knowledge of Route Conditions and Shipment Status [§73.25(b)(1)(ii) and (iii)]

Intent

The provision for updating route information is to ensure that the route selection process uses the most up-to-date information available to avoid placing the shipment on a course with unusual hazards to shipment security. After the shipment has begun to move, further updating of route information ensures that development of hazardous conditions that could affect shipment security is discovered soon enough that itinerary changes may be considered to allow the security organization to avoid, or prepare for coping with, such conditions. The provision for updating knowledge of shipment status is intended to ensure that the movement control center becomes aware of any unusual conditions encountered by the escort force during the shipment. This provision is also designed to allow a full understanding of reasons for any disruption in communications capability with the transport and its escort force because of emergency or equipment failure. Detailed guidance to response forces on where to begin looking for a missing SSNM shipment is also to be provided.

Subsystems and procedures for maintaining knowledge of route conditions as the shipment progresses may include elements independent of the escort force itself to ensure that collection of such information is obtained in a timely fashion, despite other duties escorts will have to perform during the shipment. Such procedures may include contacts with LLEAs, local news media, wire services, or weather services. Also, in the case of road shipments, escorts may be dispatched ahead of the transport to ascertain that upcoming rest or fuel stations, transfer points, or other critical locations are free of suspicious activity, civil disorders, or other threatening conditions for which special precautions must be taken or preparations made.

The physical protection system may include procedures for the movement control center to obtain shipment position and status information from the escort force on a prescheduled basis that has been mutually agreed on. This will allow determining as soon as possible when difficulties that prevent direct communications arise. Such procedures may involve, for example, check-in reports by the escort force at periodic time intervals or at predetermined milestones such as at all transport stops, at transfer points, or on crossing over from one route segment to another. These procedures may also incorporate a method for the movement control center to determine whether updating information they receive on the status of the shipment is authentic or is provided under duress. Also, various means may be employed to ensure that the movement control center becomes aware of when the updating information on the shipment status becomes critically overdue.

The following sections in the reference system apply to the suggested capability for updating the information on the status of the shipment: paragraph 73.26(i)(6) for road shipments, paragraph 73.26(k)(4) for rail shipments, and paragraph 73.26(l)(7) for sea shipments. Note that although there is no suggestion for periodic check-in reports during the air segments of a shipment, the movement control center must still possess the capability for maintaining knowledge of the status of air shipments. This function, however, may be performed by periodic check-ins or, independently of the shipment, by the movement control center's monitoring flight status through airport authorities to determine if the flight is overdue or still on course. Also, check-in reports by the escort force immediately before takeoff and just after landing could narrowly circumscribe when the aircraft is in actual flight.

Content

Describe subsystems and procedures employed by the movement control center to maintain knowledge of route conditions from the time shortly before the shipment begins movement through the duration of the shipment and to determine the significance of such information for the continued safety and security of the shipment. This description should include the means employed to obtain information from diverse sources and to filter it to determine its significance, as well as the means to store and retrieve previously obtained information, if applicable. Describe procedures employed by members of the escort force to obtain, while en route, information regarding upcoming route conditions.

Describe any means used to determine information source authenticity and whether information obtained directly from the escort force is being provided under duress. Also describe the process for ensuring that appropriate action will be taken if the required information is not obtained in a timely manner (i.e., according to preplanned schedule). Refer, if necessary, to the Responsibility Matrix and other parts of the Safeguards Contingency Plan.

2.3 Determination of Alternative Itineraries [§73.25(b)(1)(iv)]

Intent

During a shipment, events that would affect the shipment's security may occur, were the shipment to continue upon its predetermined itinerary. In such cases, prudence may dictate certain changes in the schedule or itinerary that would decrease the vulnerability of the shipment. This provision is designed to ensure that a capability to rapidly and intelligently determine and communicate such changes is established in advance. It would probably be most desirable if such changes were made within the context of alternative route plans already prepared for use in such contingencies and already familiar to the escort force and the movement control center. To ensure the continuity of the high level of protection provided the shipment, these procedures also will usually include contacting LLEAs to inform them of any changes in shipment schedule or itinerary that would affect them.

Content

Describe procedures to determine and implement alternative itineraries when such changes become necessary. Describe the process by which the decision to change an itinerary or schedule is made. Specify who has final authority to make such decisions. Refer, if desired, to the appropriate sections of the Safeguards Contingency Plan.

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Chapter 3 DETECTION AND DELAY OF UNAUTHORIZED ACCESS OR MATERIALS INTRODUCTION BY STEALTH OR FORCE

These sections cover detection and delay of unauthorized access or introduction of materials into the vicinity of SSNM or SSNM transports by stealth or force when the transport is either moving or stationary.

3.1 Establishment of Controlled Access Areas [§73.25(b)(2)]

Intent

These provisions are designed to set up a restricted area surrounding the transport or the SSNM (when located outside the transport or aboard moving transports) in order to isolate the transport or SSNM and decrease the number of personnel, materials, equipment, and vehicles allowed to come in contact with the transport or the SSNM. This controlled access area simplifies preventing unauthorized access of persons or materials into the transport or into the immediate vicinity of the SSNM. It ensures that only persons or materials required to have such access to perform essential functions are in the area at any given time. Unauthorized persons would be discouraged from attempts at penetrating the controlled access area and normally would be detained at its perimeter. Establishing controlled access areas would have the following effects: (1) it would limit opportunities for attempted unauthorized access to the transport or SSNM by stealth, (2) it would increase the probability of detection of persons possessing significant amounts of tools, explosives, or other unauthorized materials useful for purposes of committing radiological sabotage or of gaining unauthorized access to the SSNM, and (3) it would ensure that attempts to penetrate the transport or to misappropriate SSNM by force will be resisted sooner with more chance of preventing theft or sabotage.

Controlled access areas are defined in paragraph 73.2(z) in terms of two basic attributes: (1) there must be a means of demarcating the area's boundaries that is clear to both authorized and unauthorized personnel and (2) there must be some way of controlling access to the area at all times to ensure that unauthorized personnel are not admitted.

The plan should address the establishment of controlled access areas, where applicable, in two somewhat different types of situations, i.e., both at transport stops and aboard moving transports. Some types of transports such as certain aircraft, rail cars, and road transports may not need to be protected by the establishment of a controlled access area while they are in motion. This is because the cargo compartments of such transports are not normally accessible to persons on board the moving transport and, further, would be inaccessible to persons on the ground by virtue of the transport's motion. Special provisions for protection may still be required in instances where the transport slows sufficiently to allow access (e.g., slow-moving trains, road transports at toll booths or traffic signals). Aboard trains or large ocean-going vessels where unauthorized persons may possibly gain access to the SSNM, controlled access areas may need to be established. A secured cargo compartment may serve as a controlled access area on board such moving transports.

In considering the space to be included in a controlled access area aboard a moving transport, the precise object to be considered the "transport" may vary according to the mode of transportation involved. "Transport," as defined in paragraph 73.2(cc), includes any vehicle for "land, sea, or air conveyance or modules for these conveyances such as rail cars or standardized cargo containers." In the case of dedicated truck or aircraft, the entire vehicle can be considered the transport. Sometimes, a controlled access area must be established on board a nondedicated transport in order to isolate the SSNM from unauthorized persons occupying the remainder of the space on the transport (if it is physically possible for such persons to gain access to the SSNM while the transport is in motion).

In the cases of large oceangoing vessels or a long train of rail cars, isolation of the entire conveyance may be impractical. Instead, aboard a specially designed ship (container ship), the standard multimode shipping container (which alternately serves as both a truck trailer or rail car mountable container) may be considered the transport. For rail transportation, the particular rail car containing the SSNM may be considered the transport. In these cases, the function of the controlled access area is to isolate the transport module (i.e., the rail car or standard shipping container) from the remainder of the conveyance of which it is a part. Alternatively, a secured cargo compartment may serve as a controlled access area aboard a moving transport.

Access detection subsystems and procedures are required to detect penetrations of controlled access areas by unauthorized persons or materials. Detection is normally expected to occur at the time of penetration, which would permit a response sufficiently timely and effective to prevent the penetration (or attempted penetration) from resulting in the theft of SSNM and limit the opportunity for a potential adversary to complete actions of radiological sabotage. Of course, the considerations that persons or materials may already have been introduced aboard a moving transport for the purposes of sabotaging the transport, and that it may not be practical for the escorts to maintain effective control over the entire transport, limit the chances for protecting against sabotage to the entire transport. However, the physical protection system may effectively protect against actions of force and stealth aimed specifically at the SSNM or the transport module in which the SSNM is contained. This protection would be designed to limit the opportunities for a potential adversary to cause significant dispersals of SSNM by direct attack on the transport or on the SSNM through actions of stealth or force.

Detection of unauthorized penetrations of controlled access areas at transport stops will normally be accomplished by escorts who will maintain constant surveillance over the SSNM or transport. For longer periods such as overnight storage, intrusion alarms may be useful. Storage of the shipment within a protected area maintained at a fixed site by an NRC licensee would satisfy the requirements for a controlled access area. The reference system delineates several safeguards measures that usually will be included in the physical protection system to satisfy the performance capability requirements for the detection of unauthorized penetrations of controlled access areas at transport stops and transfer points [paragraphs 73.26(f)(2), (i)(7), (j)(3), (k)(1), and (l)(4)].

Aboard moving transports, either guard surveillance (either direct or by using closed circuit television (CCTV)) or intrusion alarms may be used to detect unauthorized penetrations of the controlled access area. If a secured cargo compartment or a standard cargo container is used and is determined to be generally not accessible to unauthorized personnel (e.g., due to other containers stacked around and above it), a procedure for the use of tamper-indicating seals and inspections may be a suitable means of access detection provided the SSNM is placed under constant surveillance whenever access by unauthorized persons becomes possible. Locks and seals on the containers would normally be examined at these times. (Locks and seals on individual packages or containers of SSNM could also be used to establish continued shipment integrity, should the shipping container seals be damaged in transit.)

Communication and assessment functions associated with the access detection subsystems and procedures for controlled access areas should be capable of facilitating an appropriate response to detect penetration or attempted penetration of a controlled access area. This response is to be capable of meeting the threat described in paragraph 73.1(a). The details of the performance of these communications and assessment functions are to be discussed in the Safeguards Contingency Plan in the context of the overall response.

Content

1. Controlled Access Areas at Transport Stops. Describe the different types of controlled access areas that will need to be established at transport stops, and describe the criteria to be used for selection of appropriate locations for such stops. Include those for rest stops and emergency stops for road transportation, intermediate ports of call for sea shipments, transfer points, etc. (Refer to the Safeguards Contingency Plan as necessary.) For each type of controlled access area described, include the transportation modes involved and the occasion for the transport stop being made. Briefly describe significant adverse conditions under which the controlled access area will be expected to function (e.g., crowds, traffic, availability of nearby cover).

Provide for each of the different types of controlled access areas described, in combination or separately as appropriate, the following information:

a. Demarcation of Controlled Access Areas. Describe how each controlled access area is to be demarcated in order to indicate to escort personnel and to other authorized and unauthorized persons that a particular area has been defined and that such area is restricted to authorized personnel only. Some examples include cordoning off the area with posted guards or erecting temporary barricades.

b. Implementation Procedures. Describe procedures and criteria for escort personnel to select appropriate locations and arrangements (to the extent possible) for establishing controlled access areas. Describe procedures to be followed in establishing such areas with minimum delay when the specified occasion arises. The configuration and approximate dimensions of the controlled access area to be established in relationship to the transport or the SSNM should be included in this description.

c. Access Control. Describe the barrier subsystems, procedures, or other devices employed to ensure that only authorized personnel will be allowed access to the controlled access area. These may include posted guards (escort personnel) responsible for detaining unauthorized persons or materials at the boundary of the controlled access area, appropriate physical barriers, or other means for delaying access to the controlled access area.

d. Access Detection. Describe subsystems and procedures employed to detect and communicate any unauthorized penetration (or such attempts) of the controlled access area by persons, vehicles, or materials and to whom such detections would be communicated. Refer to the Responsibility Matrix of the Safeguards Contingency Plan as necessary.

Discuss the interrelationship among the access control and detection subsystems and the procedures described above and the armed response capabilities in order to demonstrate that the combined systems can satisfy the general performance requirements of paragraph 73.20(a).

2. Controlled Access Areas Aboard Moving Transports. Provide descriptions of the types of controlled access areas to be established aboard moving transports. Include transportation mode involved, location of SSNM aboard the transport, and access points where unauthorized persons could gain access or unauthorized materials could be introduced into the vicinity of the SSNM in the absence of appropriate controls.

Provide for each of the types of controlled access areas to be established aboard moving transports, in combination or separately as appropriate, the following information, similar to that included in descriptions of controlled access areas at transport stops: (a) demarcation of controlled access areas, (b) implementation procedures, (c) access control, and (d) access detection.

Discuss the interrelationships among the access control and detection subsystems and procedures described above and the escort force (or other armed response personnel) response capabilities in order to demonstrate that the combined systems can satisfy the general performance requirements of paragraph 73.20(a). Refer to the Safeguards Contingency Plan as necessary.

3.2 Transport Features to Delay Access
[§73.25(c)(2)(i)]

Intent

This provision is intended to ensure that the physical protection system takes into account in its overall scheme the degree of protection afforded by transport design features. Where possible, it is also intended to suggest the inclusion of design features in special-purpose transport vehicles that maximize the delay an adversary might encounter in attempting unauthorized penetration of the transport.

Generally, only road transport vehicles will be specifically designed to include features that delay access to the cargo compartment in the event of attempted penetration through stealth or force. Such features in road transport vehicles are provided for in the reference system [paragraph 73.26(i)(3)]. However, licensees may also rely on the design features of other types of transports in the design of the overall physical protection system. Some examples are the structural strength of a particular type of rail car or affixing certain equipment to or temporarily altering the transport to improve its resistance to penetration. Additional locks may be installed. All such delay features may be considered as transport features relevant to this provision to the extent that they are specifically intended or relied on to delay the adversary's attempt at transport penetration.

Content

Describe transport(s) used to carry the SSNM. Describe design features affecting transport penetration resistance. State whether transportation vehicles or modules will be under the control of the licensee (owned, leased, or rented) or under the control of a contract or common carrier. Include drawings and specifications that describe the penetration-resistant features of the portion of the transport used to carry the SSNM (e.g., the means of securing access points and the construction of the transport's body shell). Penetration from all accessible points of attack, including the top and undersides of transports, should be considered. (See Regulatory Guide 5.31, "Specially Designed Vehicle with Armed Guards for Road Shipment of Special Nuclear Material.")

3.3 Access Detection for Transports [§73.25(c)(2)(ii) and (c)(2)(iii)]

Intent

These provisions were designed to ensure that a detection capability that applies strictly to the transport, in addition to the detection capability associated with the controlled access area that may surround the transport, is provided in the physical protection system. This is to take into account that certain persons who may be authorized to enter the controlled access area may not be authorized to enter the transport itself. Some examples include transport operating crew members on ships, airplanes, and trains or transport service or maintenance personnel at stops.

The smaller perimeter of the transport, in comparison with the controlled access area surrounding it, and the usually better-defined boundaries of the transport make practical a wider variety of access detection subsystems and procedures than might ordinarily be effectively employed in the case of a controlled access area.

Access detection subsystems and procedures are needed under these provisions for many reasons. Unauthorized tampering with transports and cargo containers can be detected. Any unauthorized presence of persons or materials and any unauthorized attempt to penetrate the transport can be detected, assessed, and communicated so that the response can prevent SSNM theft. Tamper seals and inspections aid detection of tampering. Intrusion alarm systems and direct or remote (e.g., CCTV) surveillance aid detection of the actual presence of unauthorized persons within or adjacent to the transport or of attempts at unauthorized penetration. Seals do not detect SSNM theft during the act but do provide evidence that penetration has or has not occurred in the interims between inspections and can provide a useful record for tracing lost shipments should some SSNM be found missing. This nonpenetration assurance is also useful for eliminating the need to undertake more detailed inspection of transports in order to establish that there are no unauthorized persons or materials aboard a transport before loading if the transport has been previously inspected and sealed.

Detection of unauthorized access to controlled access areas is generally limited to surveillance by escorts; however, remotely or locally annunciated intrusion alarm systems become practical and feasible for transports. The same individuals (escorts) may be made responsible for detecting unauthorized access to both the controlled access area and the transport contained within it. However, this provision makes the licensee responsible for demonstrating that these two functions can be adequately performed by the same personnel simultaneously. The licensee, for example, would have to demonstrate that personnel within the controlled access area would not be likely to remain undetected in their attempts to covertly gain unauthorized access to the transport while the escorts were busy watching persons outside the controlled access area.

Content

1. Tamper Indication [§73.25(c)(2)(ii)]. Describe the inspection and detection subsystems and procedures to detect unauthorized tampering with transports and cargo containers during transit and before loading. Include schedules for inspections and tamper-indicating features of transport and physical protection hardware. Acceptable devices and procedures for satisfying these provisions are contained in Regulatory Guide 5.15, "Security Seals for the Protection and Control of Special Nuclear Material."

2. Transport Access Detection [§73.25(c)(2)(iii)]. Describe the surveillance subsystems and procedures that will be used to detect, assess, and communicate any unauthorized presence of persons or materials within the transport and any unauthorized attempt to penetrate the transport. Also describe those that ensure that no unauthorized persons or materials are on board the transport before SSNM loading or on the escort vehicles (for road shipments) immediately before the trip begins. Specify which subsystems and procedures are to be used under given conditions and circumstances to ensure that a surveillance or detection capability exists at all times during shipment. If an alarm system is used, describe its tamper-resistant features and the location and means of annunciation. Relate the response to any annunciation of an alarm system or to any alarm transmitted by security personnel to the predetermined response indicated in the Safeguards Contingency Plan.

Discuss the interrelationships and interdependencies of the transport access detection subsystems and procedures described here with other parts of the in-transit physical protection system, including the Safeguards Contingency Plan. Demonstrate how it will be ensured that detection of unauthorized access to or activities within the transport will allow a response timely and effective enough to prevent the unauthorized penetration or activity from resulting in SSNM theft.

Chapter 4 DETECTION OF UNAUTHORIZED ACCESS OR MATERIALS INTRODUCTION BY DECEIT

Access detection and control subsystems and procedures for protection against actions of stealth and force were described in Chapter 3 of the Physical Protection Plan (see Chapter 3 of Part I of this guide). They were designed to channel all materials and persons seeking introduction into controlled access areas or transports through established entry control points. The subsystems and procedures described in this chapter are designed to permit the screening of all persons and materials being introduced through such entry control points in order to determine whether they are duly authorized for such access.

4.1 Access Authorizations
[§73.25(b)(3)(i) and (c)(1)(i)]

Intent

Access authorizations should be maintained and used with access control systems for controlled access areas and transports. Their purpose is to determine the persons who are authorized to be admitted to the controlled access area or transport and the criteria for their admittance. Access authorizations in the form of a written access authorization schedule enable escort personnel to quickly discriminate between those who are to be admitted to the controlled access area or transport at any given time and those who are to be excluded. Such a schedule may also provide information that can be used to verify the identities of those persons seeking access or of those materials, equipment, or vehicles presented for introduction.

Access authorizations would normally name the individual to be granted access and provide the criteria (i.e., the time, place, circumstances, etc.) for admittance. The authorizations would be limited to the specific times when and places where the individual has the need to be admitted to the controlled access areas or transport to perform his/her job. They may also specify the activities in which the individual would be authorized to engage while enjoying such access. The authorizations would normally be provided in written form and made available to the escort force personnel who would be responsible for implementing the access control system.

In some situations, the security organization may authorize an individual access to controlled access areas or the transport on a temporary basis without specifying the individual's name or means of personal identification. This may be necessary to facilitate refueling or vehicle service at stops. Such individuals would normally be given escorts or placed under close surveillance to ensure that they perform only the tasks for which they have been specifically authorized.

These provisions also apply to materials being introduced into controlled access areas or transports. Security-significant equipment, materials, or vehicles that may potentially have an adverse impact on the security of the shipment or that may be used to conceal such equipment or materials may be excluded unless specifically authorized for introduction into a controlled access area or transport.

It is the licensee's responsibility to determine which individuals, materials, equipment, or vehicles should be given authorizations and to instruct accordingly the members of the escort force having responsibility for implementing the access control system.

Content

Describe the criteria to be used for determining which persons and materials are to be approved for authorization for access to or introduction into (1) controlled access areas and (2) transports. Categorize the criteria according to the different situations and conditions expected to apply at different times during shipment.

Indicate in which cases persons granted access will require escorts or close surveillance by duly authorized personnel. Describe or refer to the escort or surveillance procedures used to the extent they are applicable.

State which individuals in the security organization (position titles and names of individuals) may approve access authorizations. Describe procedures for processing authorizations and the forms used for documenting such authorizations (e.g., authorization lists, letters of authorization).

Describe precautions taken to ensure that authorization documents will not be generated without the approval of the responsible individuals in security organization management and to ensure that they will be secured to prevent tampering or alteration. Describe procedures for ensuring that authorization lists will be constantly updated, as necessary, and that only the most current authorization lists will be used.

4.2 Access Control at Entry Control Points [§73.25(b)(3)(ii) and (c)(1)(ii)]

4.2.1 Identification and Verification

Intent

Access control subsystems and procedures must be implemented, under these provisions, to ensure that only authorized persons and materials are allowed access to or introduction into controlled access areas or transports. Usually these subsystems and procedures will include establishing entry control points, identifying authorized persons or materials, verifying such identifications, and assessing verified identifications against established authorization schedules.

Entry control points should normally be few in number (generally not more than one per controlled access area) to ensure the greatest amount of control with the limited personnel and other physical protection resources available. Their objective is to cause each individual seeking entry to the area or transport to be channelled through a screening process for identification, verification, and authorization assessment before being allowed to enter. It is desirable to ensure that the detection of any attempted introduction of unauthorized persons or materials into controlled access areas or transports will be communicated to other escort force personnel so that appropriate measures described in the Safeguards Contingency Plan may be initiated to deny such unauthorized entries. Generally this might best be accomplished by routinely detaining persons attempting entry at the entry control point. Thus the communication of detections of unauthorized persons or materials being introduced can be accomplished before notifying the individual that he will not be admitted for lack of proper identification or authorization. This will provide as much warning as possible to other members of the security organization if the persons involved are in fact adversaries and are prepared to take violent measures upon failure of their deceit mode.

One method that will usually accomplish the identification verification function is the use of a numbered picture badge identification procedure affecting all individuals who will have custody of the shipment. This procedure is described in §73.26(g)(1) of the reference system. All members of the licensee's escort force and all members of other groups of security personnel authorized to assume custody of the shipment at transfer points or at the final destination are included. Badges held by the persons who would assume custody would be compared with facsimiles of such badges provided in advance to the escort force giving up custody of the shipment to ensure that identifications are valid. Other means may be employed to identify personnel who would be granted temporary access to an area or to the transport. Temporary badges, for example, may be issued to such persons while they are in the subject area or transport. These would signify their temporary status and, if applicable, their need for an escort or other close supervision to limit their opportunities to perform unauthorized acts.

Content

Describe how entry control points are to be established for controlled access areas of the types described in Section 3.1 of the Physical Protection Plan (see Section 3.1 of Part I of this guide). Also describe the entry control points for each of the transports described in Section 3.3 of the plan (see Section 3.3 of Part I of this guide).

For each distinct type of entry control point described above, describe the subsystems and procedures used to detain persons or materials for which introduction is being sought until their identities and authorizations can be confirmed.

Discuss measures that would be taken if an attempt is detected to gain entry to introduce unauthorized vehicles or materials through authorized entry control points by deceit. Refer to the Safeguards Contingency Plan as necessary.

Describe how identification and documents presented for the purpose of identification will be verified as being authentic.

4.2.2 Assessment Against Authorizations

Intent

After a person's identity has been verified, such identifications must be assessed against a written authorization schedule. This assessment will determine that the person seeking entry has been authorized to enter the controlled access area or transport and that the specific entry criteria described in the authorization schedule have been met. The use of entry criteria is intended to ensure that access is permitted only to allow non-escort personnel to perform specific assigned duties that require access to the controlled access area or the transport. Further, the specification of certain entry criteria makes it more difficult for a potential adversary to gain access by deceit by decreasing the amount of time and the number of occasions on which adversaries could deceitfully obtain such access.

Specification of conditions that must be met for the introduction of materials places similar limitations on the potential adversary's ability to successfully introduce unauthorized materials into controlled access areas or transports by deceit.

Measures must also be taken to ensure that authorization schedules and other related access authorization items are maintained in a secure manner to ensure their continued availability and to prevent compromise of the authorization assessment subsystems and procedures.

Content

Describe subsystems and procedures used to assess verified identifications of persons, vehicles, and materials against current authorization schedules and entry criteria (conditions placed upon authorized entry related to particular times, places, and circumstances) before such persons, vehicles, or materials are permitted access or introduction. Describe details of the forms of authorization schedules that will be used by personnel or devices involved in the authorization assessment function. Also discuss the precautions taken to prevent the successful use of counterfeit authorization items.

Chapter 5 PREVENTION OF UNAUTHORIZED REMOVAL OF SSNM FROM TRANSPORTS BY DECEIT

5.1 Authorization for Removal of SSNM from Transports [§73.25(c)(3)(i)]

Intent

Authorizations are necessary under the provisions for removal of SSNM from transports at intermediate stopping points and at the final destination. The purpose of such authorizations is to ensure that the SSNM will not be withdrawn unnecessarily or indiscriminately from the transport and placed in a more vulnerable situation as a result of actions of deceit by adversaries. Authorizations for removal of SSNM from a transport are, in effect, instructions to the escort force personnel on times and conditions suitable for removing SSNM from its protected location inside the transport. When persons are presenting themselves as potential recipients of the SSNM, it is especially important to confirm that their identities are the same as those described on the authorization schedule before exposing the SSNM to them by removal from the transport.

SSNM will have to be authorized for removal from transports at some transfer points and at the final destination. Numerous scheduled removals from the transport are discouraged and can usually be avoided by use of appropriate equipment and planning. In a shipment involving road, rail, and sea segments, for example, removals of wieldable packages of SSNM from one transport for loading into another can be avoided by using a standardized shipping container of the type specified in the reference system [paragraph 73.26(l)(1)].

To ensure that SSNM is removed from transports only when necessary, authorization schedules for the removal of SSNM from transports must specify authorized times, places, and conditions for removal, plus names of persons authorized to remove and to receive the SSNM. They must also be consistent with the means used to identify the recipients and the controls and procedures for removal. (See Regulatory Guide 5.57, "Shipping and Receiving Control of Special Nuclear Material.")

Content

Describe the criteria for removal of SSNM from the transport. State the names or position titles of persons who will be authorized to implement removal procedures, as necessary.

Describe the SSNM removal authorization schedules that will be maintained to determine when prescheduled removals are to be executed. Also describe the forms of documentation (e.g., hand-to-hand receipts as suggested in the reference system, paragraph 73.26(b)(4)) that will be used to provide a permanent record of transfer of the SSNM when custody of the SSNM is transferred. Samples of authorization schedules and other documentation should be provided.

5.2 SSNM Removal Controls
[§73.25(c)(3)(ii) and (c)(3)(iii)]

Intent

Removal controls are necessary to ensure that the SSNM removal authorization schedules just discussed are appropriately implemented and that the security of the SSNM during the removal and subsequent transfer operations is adequately maintained. This is to ensure that the actual SSNM removal is undertaken only after authorization for such removal has been confirmed and the identities of persons into whose hands the SSNM is to be transferred have been verified and assessed against the current authorization schedule. In this sense, the removal controls are a type of filtering process similar to access controls in discriminating between those who are authorized to receive SSNM and those who are not.

Removal controls also include verifying the identity and integrity of the SSNM to be removed and ensuring that the removal and transfer of SSNM from the transport is properly documented in order to establish proof that responsibility for the shipment has been duly transferred to the appropriate party. (Hand-to-hand receipts are suggested in the reference system in paragraph 73.26(b)(4). See Section 1.6, "Security Records," of Part I of this guide.) This documentation will ensure the traceability of the shipment if it is missing or overdue at its final destination.

Content

1. Emergency Removal Procedures. Indicate that procedures are described in the Safeguards Contingency Plan for the removal of SSNM from the transport in emergency situations for each transport mode involved and that such procedures include how the SSNM is to be protected while outside an approved transport, what procedures are to be followed in transferring material to a substitute transport, and how the escort force would maintain control over the SSNM if a substitute transport were used. Refer to the appropriate portions of the Safeguards Contingency Plan.

2. Prescheduled Removal Procedures. Describe procedures to be used during prescheduled removals of SSNM from transports. Include procedures for identifying persons physically removing the SSNM and those who will be receiving the material (if custody of the material is being transferred). Also include procedures for verifying the identifications of such persons, verifying the identity and integrity of the SSNM that will be removed from the transports, and assessing each of the verified identifications just described against the most current authorization schedules. Describe the procedures to ensure that all the foregoing procedures will be completed satisfactorily before physical removal of SSNM from the transports.

Describe the forms for acknowledging receipt of SSNM at all points where there is a transfer of custody of the material. Describe the procedure or devices for validating the receipt of the material (e.g., signature of responsible individual accepting custody, use of corporate seal).

Chapter 6 DETECTION OF UNAUTHORIZED REMOVAL OF SSNM FROM TRANSPORTS BY STEALTH OR FORCE

6.1 Transport Features to Delay Removal [§73.25(c)(4)(i)]

Intent

The delay that may be encountered by a potential adversary attempting the unauthorized removal of SSNM from a transport by stealth or force is a significant factor to be considered in the overall design of a physical protection system for SSNM in transit. Transport features that delay removal of the SSNM must be considered, whether they have been specifically designed into a special purpose transport vehicle or are coincidentally present in a multipurpose transport selected to transport SSNM but also relied on to delay unauthorized removal of SSNM.

This provision has been included to ensure the use, where feasible, of design features in specially designed transports to increase the delay that would be encountered by an adversary attempting unauthorized penetration of the transport by stealth or force. Where specially designed transports are not feasible, these provisions would permit the physical protection system to be given credit for the transport features that promote such delay found in multipurpose transports.

Some features that delay unauthorized removal of SSNM from a transport may be identical to those that prevent or delay unauthorized access to the transport. Those features have been described previously in Section 3.3 of Part I of this guide. This section addresses only the aspects of those features that promote delay in SSNM removal from the transport.

A number of fairly simple features may be used to cause delay in unauthorized SSNM-removal attempts. These include fastening SSNM containers to the floor or other part of the transport interior, binding the containers together so that the resulting bulk is too heavy or too large to be moved out of the transport cargo compartment entry point in one action, or filling the cargo compartment of the transport with tear gas or other chemicals designed to obscure vision or otherwise make removal more difficult. However, it is important that the mechanisms used to cause such emissions be designed to prevent unintended accidental triggering of the devices.

In the case of road shipments particularly, there is an additional device that would be included under the heading of delaying removal of SSNM, but this involves removal of the entire transport rather than removal of SSNM from the transport. In the reference system, it is suggested as one of two options that the transport vehicle for road shipments be equipped with a device for immobilizing the transport or at least the cargo-carrying portion of the transport so that a potential adversary could not use the transport to depart with the shipment after attack [paragraph 73.26(i)(3)(i)]. This device is designed to delay the adversary from quickly leaving the attack scene with the SSNM before LLEAs arrive on the scene. This device should be sufficiently effective to prevent mobility from being restored to the transport by the adversary in a short period of time, thus defeating the purpose of the immobilization device. Use of a cargo vehicle that does not have this feature would suggest the need for a compensatory increase in other forms of protection such as the use of an additional escort vehicle (as suggested in the reference system).

Content

Describe the transport features that will be relied on to delay attempts at unauthorized removal of SSNM from the transport. Include details of construction or arrangements made to produce the delay and an estimate of the time and resources required to overcome the obstacle or mechanism causing the delay. Refer to the descriptions of transport features delaying access in Section 3.3 of the Physical Protection Plan (see Section 3.3 of Part I of this guide) as necessary.

Describe any transport features that will be relied on to delay removal of a road transport vehicle loaded with SSNM. Include details of construction and operation of any mechanisms designed to immobilize the vehicle and the actions or conditions required to trigger the applicable mechanisms (e.g., improper operation of the vehicle by unauthorized persons, remote-controlled radio signal activation). Include details of the extent to which the vehicle can be restored to normal operating condition following immobilization and the time, resources, and knowledge required to restore the vehicle to normal operating condition. Describe the fail-safe features that ensure that the trigger mechanism for the immobilization device does not become accidentally activated during shipment progress.

6.2 Detection of SSNM Removal Attempts [§73.25(c)(4)(ii)]

Intent

This provision is intended to provide a warning to the security organization that an attempt at unauthorized removal of SSNM from the transport is in progress or is imminent in order to enable a response that will prevent ultimate theft of SSNM.

There is a degree of redundancy between subsystems and procedures designed to detect attempts at unauthorized access to transports and those designed to detect attempts at unauthorized removal of SSNM from transports. However, these two categories of detection subsystems and procedures need not be identical. Detection subsystems and procedures not apparent to the potential adversary until penetration is complete are less easily compromised than those apparent from outside the transport. Also, whereas the physical protection system may rely on personnel to conduct surveillance of the transport to detect unauthorized penetration attempts, the detection of unauthorized removal attempts may rely on a mechanical device that cannot be compromised by those conducting the surveillance. This provides some degree of protection against both acts of coercion against the escorts conducting surveillance, as well as acts of collusion by the escort force members. Therefore, it would be desirable that there be some degree of independence between the detection subsystems and procedures designed to detect penetrations and removals.

If escort personnel responsible for monitoring access to the controlled access area surrounding a transport are also assigned responsibility for responding to detections of unauthorized penetrations of the transport, an adversary might successfully neutralize such escort personnel and gain access to the SSNM without the LLEAs having been notified. Greater effectiveness may be achieved in the physical protection system by arranging for separate detection subsystems for the controlled access areas and for the transport.

Alarms could be annunciated at a remote location occupied by a portion of the escort force. Because the act of unauthorized removal is so close to the realization of a successful theft, annunciations of the alarm could be considered ample cause for contacting and alerting the LLEAs before further action is taken by the escort force to assess the cause of the alarm. This detection system should be consistent with the Safeguards Contingency Plan and Sections 7.2 and 7.3 of the Physical Protection Plan (see Sections 7.2 and 7.3 of Part I of this guide) to ensure that the entire system will be able to detect, assess, and communicate attempts at unauthorized removal so that the response to the attempt can be such as to prevent the theft of SSNM as required in the statement of the required capability.

Content

Describe the detection subsystems and procedures to detect, assess, and communicate any attempts at unauthorized removal of SSNM from transports. Indicate the times when such systems would be in operation, especially with regard to periods during which other detection subsystems and procedures would be or would not be in operation to detect attempted unauthorized penetration of the transport. Discuss the extent to which the detection subsystems and procedures designed to detect unauthorized removal of SSNM from transports and unauthorized penetration into transports operate independently and offer a degree of redundancy to the detection function in relationship to the transport.

Discuss the effectiveness of the subsystems and procedures designed to detect unauthorized removal of SSNM from transports and their interrelationships and interdependencies with other parts of the physical protection system described in the Physical Protection Plan and in the Safeguards Contingency Plan. This discussion of interrelationships and interdependencies should demonstrate that unauthorized attempts at removal of SSNM from transports will be detected.

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Chapter 7 TRANSMISSION OF DETECTION, ASSESSMENT, AND OTHER SECURITY-RELATED INFORMATION

Various requirements for communications capabilities have been described in other portions of this Standard Format (i.e., Chapters 2, 3, 5, and 6 of Part I). The provisions for communications capabilities in this chapter are intended to support the other capabilities where communications capabilities are needed.

7.1 Communications Among Escort Force Personnel [§73.25(d)(2)(i)]

Intent

Communications are routinely required among members of the escort force at different locations, especially in the cases of road transportation where escort force personnel are dispersed among several escort vehicles or where some may purposely be sent to a remote location to ensure a response in the case of attack (see Section 1.8 of Part I of this guide). In addition, communications may be required among escort personnel while some or all of them are on foot, should this become necessary as a result of, or as a tactic in, certain response actions. Thus, communications among escort force members, independent of vehicle-mounted equipment, should normally be provided in some measure. In all cases, for any transportation modes contemplated in the plan, communications subsystems and procedures should be designed to permit proper implementation of the escort force duties and procedures detailed in the Physical Protection Plan and the Safeguards Contingency Plan. Continuous two-way intraconvoy communications are specified in the reference system in the case of road shipments [paragraph 73.26(i)(6)].

Communications among escort force members may be affected by atmospheric conditions, topographic features, and spurious or purposely generated electromagnetic radiation that may interfere with the ability to communicate clearly. The licensee should select subsystems and procedures to cope with such anticipated difficulties. Communications codes or other devices to disguise position or other security-significant information transmitted among escort force members may also be desirable.

Content

Describe the communications subsystems and procedures that will be employed to allow communications among members of the escort force, and relate their performance characteristics to the way they are expected to be used at different times during the shipment. Include range and limitations of the equipment, the extent to which the equipment is portable (e.g., portability by an individual in the field, restriction to vehicle mounting, or longevity of power source under continuous use), and procedures for disguising position data and other security information that may be of interest to the potential adversary (i.e., duress codes). Refer as necessary to other parts of the Physical Protection Plan or to the Safeguards Contingency Plan.

7.2 Communications Between the Escort Commander and the Movement Control Center [§73.25(d)(2)(ii)]

Intent

Two-way communications are required between the escort commander and the movement control center to transmit assessment information and requests for assistance from LLEAs. Equipment for such communications may be vehicle mounted but should provide for dead areas and other anticipated conditions that may affect communications from time to time. Segments of the route that present particular communications problems should be taken into consideration. Appropriate measures should be taken to ensure sufficient communications capability to support the objectives of the physical protection system. Although communications are not required on a continuous basis between the escort commander and the movement control center, even in the case of road shipments, a capability for such communications must be provided to allow either direct or indirect communication between the escort commander and the movement control center should an emergency arise and assistance from LLEAs be required. Alternatively, direct communications with the LLEAs would be acceptable in some situations when communications with the movement control center cannot be sustained.

The reference system provides that two of the vehicles used in road transportation be equipped with radiotelephones for communication with the movement control center and that there be some other redundant means of communication as well [paragraph 73.26(i)(6)]. For sea shipments, ship-to-shore communications are specified in the reference system [paragraph 73.26(l)(7)], whereas reference system provisions for air and rail shipments do not specify the type of communications system to be used.

Content

Describe the communications subsystems and procedures that will be used for two-way communications between the escort commander and the movement control center to rapidly and accurately transmit assessment information and requests for assistance by LLEA forces and to coordinate such assistance with the actions of the escort force. Include the range and limitations of the equipment, whether local relays will be required, and the means to be employed for disguising position data and other information in the communications that may be useful to the potential adversary. Regulatory Guide 5.32, "Communication with Transport Vehicles," provides information on radiotelephones and systems and procedures for disguising position data, which would be most useful for road or rail shipments.

7.3 Notification of LLEAs [§73.25(d)(2)(iii)]

Intent

Both the armed escorts and the movement control center personnel under this provision must be provided with the communications equipment and appropriate procedures to allow them to notify the LLEAs of the need for assistance. The reference system suggests specifically a need for redundancy in these communications in certain instances, as discussed in Section 1.8 of this plan. The need for such redundancy underscores the basic philosophy of the licensee physical protection system. The escort force is not intended to be able to defeat potential attackers in an aggressive mode but to protect the SSNM shipment and themselves from harm, while delaying the theft of SSNM or radiological sabotage, until LLEAs arrive. In a system designed to interface closely with LLEA forces, the ability of escorts and the movement control center to communicate with such forces is very important and therefore requires an appropriate level of redundancy.

Although the need to maintain a capability for communications with the LLEAs applies to all transportation modes, this does not necessarily mean direct communications. Various types of relays would be possible, especially where redundant modes are concerned. An illustration of how this communications capability can be ensured during transfer operations when the SSNM is probably in its most vulnerable condition is provided in the reference system, which provides that the escort force would split up into two groups made up of two escorts and five escorts during transfers. The two escorts would go to a remote location, and both the two remotely located escorts and the five who remain to protect the shipment would be able to communicate directly with the LLEAs [paragraph 73.26(f)(2)]. However, there are no further specific provisions for communications with the LLEAs in the reference system.

Content

Describe the communications subsystems and procedures for (1) the armed escorts and (2) movement control center personnel to notify LLEAs of the need for assistance. Refer, if desired, to Sections 7.1 and 7.2 of the Physical Protection Plan (see Sections 7.1 and 7.2 of Part I of this guide). Include in the description the range and limitations (e.g., dead areas for radio communications) of the communications, whether local relays will be required, and whether the escort force will have direct control of the communications system or will share the communications facilities with other persons (e.g., ship's radio on oceangoing vessel, aircraft radio normally used by aircraft crew).

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PART II

PHYSICAL PROTECTION ARRANGEMENTS FOR SPECIFIC SHIPMENTS

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1. GENERAL INFORMATION

Intent

It is provided in paragraph 73.26(b) of 10 CFR Part 73 that security arrangements for each shipment of SSNM be approved by the Nuclear Regulatory Commission prior to the time for the 7-day notice required for shipments by §73.72. The purpose of this provision is to permit the NRC to review the shipment physical protection arrangements for a specific shipment to ensure that they provide the level of protection required by §§73.20 and 73.25 in the full context of the conditions and situations existing shortly before the time of shipment. Although there is no requirement as to how soon before the shipment this information is to be submitted, it is normally assumed that it would not be so far in advance that the licensee would have to make drastic updating revisions to the submittal just before the actual shipment. The availability of scheduling information from carriers and other sources would affect the date of submittal and would discourage submittal too long before the shipment date. Significant changes in the arrangements reported to NRC should be submitted immediately to NRC as amendments to the original submittal when such changes appear necessary. These amendments will then be approved if extensive review is not indicated.

The provisions of paragraph 73.26(b)(3) of 10 CFR Part 73 define basically the types of information to be submitted for each shipment. However, a degree of flexibility is allowed in the amount of information that would be submitted. The licensee may choose to provide as much information as possible in the Physical Protection Plan so that the submittal for each specific shipment would be as little burdensome as possible. Providing details in the Physical Protection Plan, then simply referencing them in the submittal for each shipment would reduce the burden. On the other end of the spectrum, the licensee may choose to include in the submittal for a particular shipment a number of departures from the Physical Protection Plan, perhaps made necessary by temporary conditions beyond his control. This provision would be most suitable for temporary changes effective for only one shipment. There are also provisions for updating the Physical Protection Plan as well. The licensee, at his discretion, may choose to widely distribute the details of the physical protection arrangements for a particular shipment between the Physical Protection Plan and the submittal of Physical Protection Arrangements for Specific Shipments with the understanding that both together describe the physical protection system for any particular shipment. It is advantageous to keep the volume of any such submittals to a level commensurate with the time allowed for NRC review. The sections that follow provide a standard format for the different types of information that may be submitted before a particular shipment. The physical specifications for such submittals are the same as for the Physical Protection Plan, as described in the Introduction to this Standard Format.

Content

State the purpose of this submittal as supplying the NRC certain information on specific security arrangements for shipments of SSNM in formula quantities sufficiently in advance of the shipment to allow NRC approval of such arrangements before the time for the 7-day notice required by §73.72 of 10 CFR Part 73. Reference the Physical Protection Plan corresponding to this submittal by docket number and date.

2. ITINERARY INFORMATION

Content

State the estimated dates and times of departure and arrival, the names of the shipper, receiver, and carrier(s); and the mode(s) of shipment.

Describe the route to be used for the proposed shipment by referring to the appropriate section in the Physical Protection Plan, indicating any differences in the itinerary compared to that described in the Physical Protection Plan. Provide the appropriate information corresponding to such differences as required in the Physical Protection Plan so as to completely update the itinerary information required for the proposed route. Note such changes and include them as an appendix to this submittal.

3. TRANSFERS OF SHIPMENT SECURITY RESPONSIBILITY

Content

Note by reference to the stated itinerary the times and locations of points where there will be transfers of responsibility for the security of the shipment (i.e., escorts either accepting or relinquishing responsibility).

At each such point, describe the arrangements made to continuously maintain the security of the shipment during the transfer, including points at which the escorts will accept responsibility for an import shipment. Refer to the Physical Protection Plan as necessary.

4. DIFFERENCES FROM THE PHYSICAL PROTECTION PLAN

Content

Describe any subsystems or procedures to be used for the particular shipment described here that differ significantly from those described in the Physical Protection Plan. Present this information sequentially, according to the headings that apply to the subsystem or procedure to be supplanted. In the case of each difference from the Physical Protection Plan, indicate why the change is being made.

5. RELATIONSHIP TO THE PHYSICAL PROTECTION PLAN

Content

Ensure that, except as specifically noted in this submittal, all subsystems and procedures for the physical protection of the proposed shipment described here will be substantially the same as those described in the Physical Protection Plan.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 23, 1980

Regulatory Guide 5.60

REGULATORY GUIDE DISTRIBUTION LIST (DIVISION 5)

SUBJECT: Regulatory Guide 5.60, "Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material in Transit"

Regulatory Guide 5.60 was distributed for comment to all affected licensees and to other interested parties who attended the NRC Upgrade Rule Guidance Seminar held on March 27-28, 1979, in Richmond, Virginia. No comments were received as a result of this distribution, and the guide is now being issued as an active guide so that affected licensees may use it for preparation of their physical protection plans in response to the new requirements of 10 CFR Part 73 published in the Federal Register on November 28, 1979 (44 FR 68184).

Because Regulatory Guide 5.60 was not issued as a draft guide for comment, it is being provided as an active guide to all addressees on the Division 5 distribution list. Although comments are always encouraged on all regulatory guides, comments on this guide are particularly encouraged at this time.

Robert B. Minogue

Robert B. Minogue, Director
Office of Standards Development

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NUCLEAR REGULATORY COMMISSION
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DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

January 1980
Division 3
Task FP 925-5

Contact: M. S. Weinstein (301) 443-5910

NUCLEAR CRITICALITY SAFETY FOR PIPE INTERSECTIONS CONTAINING AQUEOUS SOLUTIONS OF ENRICHED URANYL NITRATE

FOR COMMENT

A. INTRODUCTION

Section 70.22, "Contents of Applications," of 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," requires that applications for a specific license to own, acquire, deliver, receive, possess, use, or transfer special nuclear material contain proposed procedures to avoid accidental conditions of criticality. This regulatory guide provides guidance for complying with this portion of the Commission's regulations by describing procedures acceptable to the NRC staff for the prevention of criticality accidents in the storage and processing of aqueous solutions of enriched uranyl nitrate in pipe intersections.

B. DISCUSSION

ANSI/ANS-8.9-1978, "Nuclear Criticality Safety Guide for Pipe Intersections Containing Aqueous Solutions of Enriched Uranyl Nitrate,"[1] was prepared by Subcommittee 8, Fissionable Materials Outside Reactors, of the Standards Committee of the American Nuclear Society. ANSI/ANS-8.9-1978 was approved by the American National Standards Committee N16, Nuclear Criticality Safety, in 1977 and subsequently by the American National Standards Institute (ANSI) on February 15, 1978.

ANSI/ANS-8.9-1978 is applicable only to the storage and processing of homogeneous aqueous solutions of uranyl nitrate containing no more than 93.5 wt-% U-235 in intersecting pipe geometries consisting of single central columns

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[1] Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525.

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by MAR 14 1980

Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

with smaller intersecting arms. The standard presents tabulated maximum safe values, derived from an empirical model based on validated calculations and experimental data, for both the central column diameter and the total area of intersection of the arms with the column for conditions of partial and total reflection. The standard also presents a general procedure to ascertain the safety of specific pipe intersections as well as guidance on model parameters and limitations. In intersecting pipe geometries where the model does not apply, e.g., multiple columns or columns in the vicinity of other fissionable materials, the standard requires that experimental data or validated computational techniques be used to determine the safe configuration.

C. REGULATORY POSITION

The nuclear criticality safety practices and guidance for calculating safe pipe intersections for enriched uranyl nitrate solutions contained in ANSI/ANS-8.9-1978 provide procedures generally acceptable to the NRC staff for the prevention of criticality accidents in the storage and processing of aqueous solutions of enriched uranyl nitrate in pipe intersections subject to the following:

1. Use of ANSI/ANS-8.9-1978 is not a substitute for detailed nuclear criticality safety analyses for specific pipe intersections.

2. Section 6 of ANSI/ANS-8.9-1978 lists additional documents referred to in the standard. The specific applicability or acceptability of these listed documents will be covered separately in other regulatory guides, where appropriate.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants regarding the NRC staff's plan for using this regulatory guide.

The proposed guide has been released to encourage public participation in its development. Except in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method to be described in the active guide reflecting public comments will be used in the evaluation of submittals for special nuclear material license applications docketed after the implementation date to be specified in the active guide. Implementation by the staff will in no case be earlier than June 1980.

DRAFT VALUE/IMPACT STATEMENT

1. THE PROPOSED ACTION

1.1 Description

Each application for a specific license for special nuclear material is required by the Commission's regulations to contain proposed procedures to avoid accidental conditions of criticality. This draft regulatory guide provides guidance for the prevention of criticality accidents in the storage and processing of aqueous solutions of enriched uranyl nitrate in pipe intersections.

1.2 Need for the Proposed Action

No guidance on nuclear criticality safety for pipe intersections has been published by the NRC. However, the NRC staff has been involved in the development, review, and approval of ANSI/ANS-8.9-1978, "Nuclear Criticality Safety Guide for Pipe Intersections Containing Aqueous Solutions of Enriched Uranyl Nitrate," which was approved by the American National Standards Institute on February 15, 1978. Guidance on nuclear criticality safety for pipe intersections should be provided, and consistent with past NRC practice, this guidance will be provided in the form of a regulatory guide endorsing the approved national standard.

1.3 Value/Impact of the Proposed Action

1.3.1 NRC

The standard presents an empirical model for calculating safe nuclear criticality parameters for systems of intersecting pipes containing enriched uranyl nitrate solution, guidance on model parameters and limitations, and a general procedure to ascertain the safety of specific pipe intersections. The proposed action establishes an NRC position on an existing national standard and therefore reduces uncertainty as to what the staff considers acceptable in the area of nuclear criticality for pipe intersections. This will result in more efficient regulatory actions and decisions and will have minimal impact.

1.3.2 Other Government Agencies

Not applicable, unless the government agency is an applicant.

1.3.3 Industry

The proposed action provides a convenient reference for the licensee or license applicant in designing nuclear criticality safety programs to meet regulatory requirements and standards of good nuclear criticality safety practice. Most of the impact on industry has already occurred during development, review, and approval of the national standard. Additional impact associated with the NRC endorsement of the standard should be minimal.

1.3.4 Workers

The proposed action should result in a more effective nuclear criticality safety program and therefore be of benefit to workers.

1.3.5 Public

A beneficial effect on the public can be expected since the proposed action encourages attainment of a sound nuclear criticality safety program.

1.4 Decision on the Proposed Action

As previously stated, guidance should be furnished on nuclear criticality safety for pipe intersections containing aqueous solution of enriched uranyl nitrate.

2. TECHNICAL APPROACH

2.1 Technical Alternatives

The proposed action is intended to provide guidance for nuclear criticality safety for pipe intersections containing aqueous solutions of enriched uranyl nitrate. Alternative approaches relating to this technical consideration are described in the following section.

2.2 Discussion and Comparison of Technical Alternatives

An acceptable approach on nuclear criticality safety for pipe intersections containing aqueous solutions of enriched uranyl nitrate can be developed based on calculational methods that have been tested, by comparison with experiment, to establish the reliability of results. As an alternative, reference can be made to ANSI/ANS-8.9-1978. An empirical model is employed in ANSI/ANS-8.9-1978 that applies to systems consisting of a main pipe (the central column) from which smaller pipes (arms) branch and uses as a criticality parameter the area of intersection of the arms with the column. Maximum safe values for both the central column diameter and the area of intersection of the arms with the column are provided for conditions of partial and full reflection. As previously stated, ANSI/ANS-8.9-1978 is an industry-accepted standard and is approved by the NRC staff.

2.3 Decision on Technical Approach

The guidance contained in ANSI/ANS-8.9-1978 will be used where applicable.

3. PROCEDURAL APPROACH

3.1 Procedural Alternatives

Potential SD procedures that may be used to promulgate the proposed action and technical approach include the following:

- Regulation
- Regulatory Guide
- ANSI Standard, endorsed by a Regulatory Guide
- NUREG Report
- Branch Position

3.2 Value/Impact of Procedural Alternatives

A NUREG is not a viable alternative because the guidance will contain positions. The matter is not of sufficient importance to justify issuance of a regulation. No Branch Position has been prepared or is anticipated. Preparation of a Regulatory Guide endorsing the ANSI standard is the only viable course of action.

3.3 Decision on Procedural Approach

A Regulatory Guide should be prepared endorsing ANSI/ANS-8.9-1978, accompanied by possible supplemental provisions.

4. STATUTORY CONSIDERATIONS

4.1 NRC Authority

Section 70.22, "Contents of applications," of 10 CFR Part 70, requires that applications for a license (to receive title to, own, acquire, deliver, receive, possess, use, or transfer special nuclear material) contain, among other information, proposed procedures to avoid accidental conditions of criticality.

4.2 Need for NEPA Assessment

The proposed action is not a major action, as defined by paragraph 51.5(a)(10) of 10 CFR Part 51, and does not require an environmental impact statement.

5. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICIES

The proposed guide is not considered as a substitute for detailed nuclear criticality safety analyses for specific pipe intersections but rather as an integral part of the analysis in order to attain a sound nuclear criticality safety program. No backfitting is expected from its issuance.

6. SUMMARY AND CONCLUSIONS

A regulatory guide should be prepared endorsing ANSI/ANS-8.9-1978, accompanied by possible supplemental provisions.

7. REFERENCES

None.

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February 1980
Division 8
Task OH 940-4

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PROPOSED REVISION 2 TO REGULATORY GUIDE 8.14
PERSONNEL NEUTRON DOSIMETERS

A. INTRODUCTION

Section 20.202, "Personnel Monitoring," of 10 CFR Part 20, "Standards for Protection Against Radiation," requires that licensees supply personnel monitoring equipment and require its use by specified individuals. This guide provides guidance acceptable to the NRC staff on the use of personnel neutron dosimeters where exposure to neutrons occurs.

B. DISCUSSION

The American National Standards Institute has approved a standard entitled "Personnel Neutron Dosimeters (Neutron Energies Less than 20 MeV)" and designated ANSI N319-1976.[1] This standard gives performance criteria, use factors, and dosimetry system calibration criteria for neutron dosimetry systems. Accuracy criteria are not included among the performance criteria, however.

This guide supplements the standard by adding an accuracy requirement. The guide also provides substitutes for certain of the standard's performance requirements because data from tests performed by both Battelle Pacific Northwest Laboratory and Rensselaer Polytechnic Institute at light-water nuclear power reactors showed that the requirements of the standard could not be met by NTA film in some circumstances. *

FOR COMMENT

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[1] Copies may be obtained from the American National Standards Institute, Inc., 1430 Broadway, New York, New York 10018.

* Lines indicate substantive changes from Revision 1 to this guide.

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by MAY 19, 1980.

Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

One problem in neutron dosimetry is that some neutron dosimeters, with the exception of albedo dosimeters, are less sensitive than most gamma dosimeters. Section 20.202 of the NRC regulations requires personnel monitoring if a worker is likely to receive a whole body dose of more than about 300 mrems in a quarter year. Some neutron dosimeters have difficulty measuring neutron doses of this magnitude, depending on the neutron spectral distribution. The difficulty becomes even more severe in a mixed radiation field where neutrons contribute only a portion of the 300 mrems. Thus, the licensees may require personnel monitoring when the neutron dose is less than 300 mrems in a quarter, but the standard requires the dosimeters to be able to detect only 300 millirems per quarter or 300 mrems divided by the number of dosimetry periods per quarter (paragraph 4.1 of the standard). This problem is discussed in Regulatory Position C.1.

C. REGULATORY POSITION

ANSI N319-1976, "Personnel Neutron Dosimeters (Neutron Energies Less than 20 MeV),"[1] provides guidance acceptable to the NRC staff on the use of personne neutron dosimeters, as supplemented and modified below.

1. PERSONNEL NEUTRON DOSIMETRY TECHNIQUES

Neutron dosimeters should be worn whenever the neutron dose equivalent is likely to exceed 300 mrems in a quarter (the minimum sensitivity required of a neutron dosimeter in paragraph 4.1 of Section 4, "Dosimetry System Performance," of the standard). If personnel monitoring is required by § 20.202 because of the total radiation exposure likely to occur but the neutron dose equivalent is not likely to exceed 300 mrems in a quarter, the following alternatives are acceptable with regard to personnel neutron dosimetry:

a. Using more sensitive dosimeters. The licensee may use a dosimeter with a demonstrated sensitivity greater than that required in the standard. Albedo neutron dosimeters are generally believed to be more sensitive than required by the standard for most neutron exposure circumstances. NTA film is sometimes used as a neutron dosimeter; however, it is not an acceptable neutron dosimeter for use at power reactors since it has been demonstrated that the major portion of neutron dose equivalent from nuclear power reactor neutrons is from neutrons whose energies are less than the threshold energy of NTA film (about 0.7 MeV). NTA film may be used as a dosimeter for other activities involving neutrons if (1) humidity can be controlled, (2) it can be shown that the energy distribution would provide a significant fraction of the dose from neutrons with energies greater than 0.7 MeV, and (3) tracks are counted on a large area of film.

b. Calculated neutron dose equivalent to supplement neutron dosimeter. A licensee may use a personnel neutron dosimeter but may substitute a calculated neutron dose equivalent for the measured dose equivalent if the measured dose equivalent cannot be reliably determined because of the lack of sensitivity of the dosimeter. Calculated dose equivalents may be based on neutron dose equivalent rates as measured by rem meter survey instruments (using either the Andersson-Braun or Hankins design) and known personnel occupancy times. Another alternative to calculated dose equivalents is the use of the neutron/gamma ratio. The neutron/gamma ratio is acceptable only if it has been established by prior use of rem meter survey instruments that the neutron/gamma ratio is virtually constant (e.g., ± 50%) throughout each area that personnel may occupy. More information on determining neutron dose equivalents by neutron/gamma ratio is given in Regulatory Guide 8.4, "Direct-Reading and Indirect-Reading Pocket Dosimeters."

c. Calculated neutron dose equivalent in place of neutron dosimeter. If an individual is not likely to receive a neutron dose equivalent in excess of 100 mrems in a quarter but would still be required to be provided personnel monitoring under § 20.202 (e.g., when gamma exposure is estimated to be in excess of 300 millirems per quarter), a personnel neutron dosimeter may be omitted. The neutron dose equivalent should then be estimated by the methods in Regulatory Position C.1.b. This procedure is discussed in more detail in Regulatory Position C.3 of Regulatory Guide 8.4.

d. Neutron dose equivalent much smaller than gamma dose equivalent. If the whole body neutron dose equivalent is not likely to exceed 10% of the gamma plus x-ray dose equivalent, neutron dosimeters may be omitted and the neutron dose equivalent may be assumed equal to zero.

e. Negligible neutron dose equivalent. If the neutron dose equivalent is not likely to exceed 30 millirems per quarter, or 10 millirems per quarter for individuals under 18 years of age, neutron dosimeters may be omitted and the neutron dose equivalent assumed equal to zero. The determination that an individual is not likely to receive a neutron dose equivalent of 30 millirems per quarter should not be based on previous NTA film badge readings since NTA film is not sufficiently sensitive to detect this dose equivalent rate.

2. PERFORMANCE REQUIREMENTS

Section 4, "Dosimetry System Performance," of the standard on performance requirements for the dosimetry system should be modified and supplemented by the following:

a. Instead of paragraph 4.1 of the standard on lower limit of detection, the following should be used:

"The lower limit of detection of the dosimetry system shall not exceed 300 millirems per quarter. The quarterly lower limit of detection (LLD) in millirems at the 95% confidence level is defined as

$$LLD = 4.66S\sqrt{N}$$

where N is the number of dosimetry exchange periods in a quarter and S is the standard deviation of the normal background on the control dosimeter in millirems."

(This definition of LLD was chosen to be consistent with the NRC position previously stated in Tables 1 and 3 of Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants." The basis for the definition is given in USERDA report HASL-300, p. D-08-01.[2])

b. Instead of the 10% limit on the standard deviation in paragraph 4.4 of the standard on precision, a limit of 30% should be used.

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[2] Harley, John H., Editor, HASL Procedures Manual, HASL-300, USERDA, revised annually.

c. The following accuracy requirement should be added: "When exposed to an unmoderated californium-252 source, the average accuracy of a set of 10 dosimeters exposed in the range from 100 mrems to 3 rems should be ±50%."

d. The tests necessary to verify that the dosimetry system meets the requirements in the standard may be performed by the licensee or by someone selected by the licensee. These tests need be performed on a system just once but should be repeated any time changes are made in the dosimetry system (e.g., processing the dosimeters differently). In the case of NTA film, each different type of packaging should be considered a different system to be tested separately. A licensee following the recommendations of this guide should maintain records to show that his neutron dosimeters have been tested and meet the performance requirements of Section 4 of the standard.

3. MEETING THE FADING REQUIREMENTS OF THE STANDARD

To meet the fading requirements in paragraphs 5.3 and 5.4 of Section 5, "Use Factors," of the standard, the licensee should irradiate four dosimeters with a neutron dose equivalent of at least 0.5 rem. The dosimeters should then be stored for an entire dosimeter exchange period at a humidity similar to that expected in normal use. The dosimeters should then be read to verify that, on the average, they meet the fading criterion (paragraph 5.3 of the standard). If humidity is a significant cause of fading, the test should be performed during a period of high humidity.

4. DOSIMETRY SYSTEM CALIBRATION

Section 6, "Dosimetry System Calibration," of the standard gives suggestions on dosimetry system calibration. The calibration may be performed by the licensee or by someone selected by the licensee, but records describing the dosimeter calibration should be maintained by the licensee. The calibration should be repeated any time the licensee has reason to believe the neutron spectrum has changed and the previous calibration might not be valid.

Calibration of dosimeter response may be performed by exposing the dosimeter and a reference monitoring instrument in the actual locations where significant neutron exposure occurs. The spherical Hankins or the Andersson-Braun design rem meters or similar instruments may be used as reference monitoring instruments. These monitoring instruments should be calibrated by sources whose calibration is traceable to the National Bureau of Standards.

Calibrations of dosimeter response may also be performed by the dosimeter processor. The processor should use a calibration factor applicable to the dose spectrum delivered to the individuals wearing the dosimeters.

5. MAINTAINING BODY CONTACT

Albedo neutron systems usually require close body contact at all times during usage. Sizable errors can occur if close body contact is not maintained. Albedo neutron dosimeters should have a means of maintaining this necessary close contact with the body.

D. IMPLEMENTATION

This proposed Revision 2 of the guide has been released to encourage public participation in its development. Except in those cases in which an applicant has proposed an acceptable alternative method for complying with specified portions of the Commission's regulations, the methods described in the currently active guide (Revision 1, August 1977) will continue to be used in the evaluation of license applications that were docketed after November 1, 1977. The methods to be described in the active Revision 2 reflecting public comments will be used in the evaluation of all license applications docketed after the implementation date to be specified. The implementation date will in no case be earlier than June 1, 1980.

DRAFT VALUE/IMPACT STATEMENT

1. THE PROPOSED ACTION

1.1 Description of the Proposed Action

Section 20.202 "Personnel Monitoring," of 10 CFR Part 20, "Standards for Protection Against Radiation," requires that licensees supply personnel monitoring equipment and require its use by specified individuals. Some workers at NRC-licensed facilities may occupy restricted areas where the potential for exposure to neutrons would require the use of personnel neutron dosimeters. The proposed action will update earlier guidance on acceptable personnel dosimetry methods for use in assessing occupational exposure to neutrons.

1.2 Need for the Proposed Action

In August 1977, the NRC published Regulatory Guide 8.14, "Personnel Neutron Dosimeters," which indicated when personnel neutron dosimeters were required by 10 CFR Part 20 to be worn and what alternative dosimetry methodologies were acceptable in establishing the neutron dose of a worker.

Recent data obtained in conjunction with two research contracts* funded by the NRC show that (1) NTA film is not a reliable dosimeter for use in assessing neutron doses when a large proportion of the neutron dose is from neutrons with energies less than 0.7 MeV, which is considered the threshold energy of film and (2) the use of a neutron/gamma ratio does not always enable accurate estimation of the neutron dose unless it can be established by routine and periodic surveys that the ratio remains constant. The scope of these research contracts was to measure and characterize the neutron spectra in the containment of pressurized water reactors (PWRs) to test the response of NTA film to the neutron spectra in numerous reactor containment structures while at power and to evaluate the response of numerous other personnel dosimeters and survey instruments where appropriate.

Owing to many causal factors such as humidity, fading, lack of sensitivity, and the general inability of NTA film to respond to neutrons with energies less than 0.7 MeV, NTA film has been demonstrated to be a very poor dosimeter for use at power reactors. Data compiled in the above research programs showed that NTA film did not respond at all or measured only a very small fraction of the neutron dose. Also, an assessment of the neutron/gamma ratio throughout a power reactor containment by one researcher (Rensselaer Polytechnic Institute) showed that the ratio may vary significantly depending on the location in the reactor containment structure and the energy response function of the type of neutron survey instrument used.

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*Rensselaer Polytechnic Institute Research Project No. 5-24675 and Department of Energy Contract EY-76-C-06-1830 with Battelle Pacific Northwest Laboratory.

Active Regulatory Guide 8.14 (Revision 1, August 1977), although it cautions the licensee of the limitations of NTA film, does not advise the licensee that NTA film is not acceptable for neutron personnel dosimetry at power reactors. The data compiled in the two NRC research programs on this subject show that technical advances in albedo dosimeters make them a better alternative to NTA film as a neutron personnel dosimeter for use at nuclear power reactors. Also, the current guide allows the use of neutron/gamma ratios as an acceptable method of calculating the neutron dose in those cases in which a dosimeter has demonstrated lack of sensitivity or the neutron dose is not expected to exceed 300 millirems per quarter but personnel dosimetry is required by § 20.202 for other forms of radiation. However, the neutron gamma/ratio varies widely throughout the containment structure of a nuclear power plant and in order to obtain accurate estimation of a neutron dose received, evidence that the ratio remains constant should be established by routine and periodic surveys of areas for which this ratio is to be applied. The NRC staff believes it is therefore appropriate to revise Regulatory Guide 8.14 to reflect this recently acquired information on neutron dosimetry.

1.3 Value/Impact of the Proposed Action

1.3.1 NRC Operations

The value in revising Regulatory Guide 8.14 is that it would provide the licensee and the NRC inspector with "state of the technology" neutron dosimetry, which has changed somewhat since the development of the existing guide. The actual development of the revised Regulatory Guide 8.14 will require few additional resources of either man-hours or funding. Two NRC contracts on analysis of neutron spectra and various dosimeter responses to moderated fission neutrons in power reactors have already been funded, and final reports will likely be completed this year. The data thus far compiled from these contracts provide the bases for the proposed changes in Regulatory Guide 8.14.

1.3.2 Other Government Agencies

Not applicable.

1.3.3 Industry

There should be little impact on industry. The revision of Regulatory Guide 8.14 will require few additional expenditures of time or resources since the revised guide is intended to reflect the current "state of technology" in personnel neutron dosimetry. The only significant economic impact that could result would be the loss of revenue by a few dosimetry processors that now provide NTA film badge service to nuclear power reactor licensees since the major purpose of revising Regulatory Guide 8.14 is to advise power reactor licensees that NTA film should not be used for neutron dosimetry.

Power reactor licensees who calculate the neutron dose by the neutron/gamma ratio method will now be required to demonstrate that the neutron/gamma ratio is constant within ± 50% in those areas of the reactor containment where it is intended to use the ratio to calculate the neutron dose. A change to the stay-time method may require the purchase of additional neutron survey instruments. However, the proposed guide states that the use of the neutron/gamma ratio method may still be justified for use as long as the licensee can demonstrate that the ratio is virtually constant throughout areas normally occupied by workers.

A recent survey of some NRC licensees by the NRC Office of Inspection and Enforcement provides data indicating that only one out of fifteen PWRs surveyed used NTA film to assess neutron dose. Also, only one PWR used the neutron/gamma ratio method instead of the stay-time method to assess neutron dose for containment entry while at power.

1.3.4 Workers

There would be no expected detrimental impact to workers. The value would be that a worker would know that "state-of-the-technology" dosimetry techniques are being examined to ensure that the best methods are being used to assess his or her exposure to neutrons. By ensuring that the best dosimetry techniques are used, the integrated risk to a worker from exposure to radiation can be better evaluated. Particular problems with NTA film include limited sensitivity, fading due to humidity, and energy dependence; these factors are often interpreted to mean a zero dose even though use of portable survey instruments has established that neutrons exist in the working area. Similarly, the improper use of the neutron/gamma ratio method may result in a poor evaluation of the worker's neutron dose equivalent.

If the worker's neutron exposure is determined using more sensitive and accurate dosimetry techniques, it may result in the limitation (i.e., reduction) of the exposure to other forms of radiation. To this extent, the risk to an individual worker could possibly be reduced. However, by using dosimeters that are more sensitive than NTA film, one might also witness an apparent increase in the individual or collective dose because neutron exposure that had previously been assumed to be zero could now be assessed. However the potential impact on the collective dose would not be great since neutron exposures at nuclear power reactors constitute a small fraction of the total collective dose. Also, only a limited number of workers who may be required to enter containment during power operation are exposed to neutrons.

The importance of very accurately assessing any exposure to intermediate or fast neutrons is highlighted by the recent controversy over the need to adjust the quality factor used to determine the neutron dose equivalent. Dr. Harold Rossi has proposed on the basis of a dual-action theory (dose squared response of low LET radiation) that the quality factor for neutrons needs to be increased an order of magnitude. If Rossi's contentions are proven correct, low levels of neutron exposure to the limited number of workers who routinely enter the containment structure of PWRs while they are at power may become the limiting exposure circumstances for these workers.

1.3.5 Public

Not applicable.

1.4 Decision on the Proposed Action

It is recommended that a revision of Regulatory Guide 8.14 be undertaken in order to ensure that nuclear power plant licensees use personnel neutron dosimetry methods that are both accurate and of appropriate sensitivity to evaluate neutron exposures experienced in working areas.

2. TECHNICAL APPROACH

Since the intent is to revise an existing regulatory guide in order to incorporate updated technical information on acceptable personnel neutron dosimeters and acceptable methods for calculating neutron doses when the use of a personnel dosimeter is not warranted, no alternative technical approaches have been identified.

3. PROCEDURAL APPROACH

The only procedural approach considered is the revision of Regulatory Guide 8.14, "Personnel Neutron Dosimeters."

4. STATUTORY CONSIDERATIONS

4.1 NRC Authority

The regulatory guide does not have the force of law, however, it is intended as an aid and describes methods that are acceptable to the NRC staff and that a licensee may adopt in order to be in compliance with a regulation that has the force of law.

4.2 Need for NEPA Statement

The revision of Regulatory Guide 8.14 has no environmental impact; therefore no NEPA statement is required.

5. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICIES

No known conflict with any existing regulation or other government policy exists.

6. SUMMARY AND CONCLUSIONS

Some data obtained from the two NRC research contracts have shown that the energy and concomitant dose distribution of fission neutrons within PWR containments is such that NTA film is of little value as a personnel dosimeter. There is little neutron exposure associated with most boiling water reactors compared with PWRs because of their design differences and because there is no entry to the drywell area during power operation of BWRs inerted with nitrogen. It is known that NTA film does not respond to neutrons with energies less than about 0.7 MeV and further that most of the neutron fluence in a typical PWR containment during power operation is less than 0.7 MeV. It is also known that NTA film is relatively insensitive to doses of less than 30 millirems per exposure and is quite susceptible to fading caused by humidity unless properly packaged. In this regard, one often finds that temperature and humidity are normally quite high in the containment of PWRs and that most containment entries to make routine inspections result in neutron doses of less than 30 mrems. Only personnel dosimeters with proven sensitivity to low- and intermediate-energy neutrons (i.e., albedo dosimeters) should be used at nuclear power plants. For low dose situations, alternative methods of calculating neutron dose may be acceptable in lieu of a personnel dosimeter. Acceptable methods include the neutron/gamma dose ratio and stay-time. However, since the neutron/gamma ratio has been shown to vary widely in work areas of PWR containments, it should be used with discretion.

In order to ensure that workers are provided with appropriate personnel neutron dosimetry, the staff has determined that Regulatory Guide 8.14 should be revised to reflect the following: (1) NTA film is not an acceptable personnel neutron dosimeter for use at nuclear power plants and (2) the use of the neutron/gamma ratio as an alternative method to calculate neutron dose is acceptable, especially for nuclear power plants, only when it is demonstrated that the ratio remains virtually constant in all occupied work areas.

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OFFICE OF STANDARDS DEVELOPMENT
DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

May 1980
Division 8
Task OH 902-1

Contact: A. K. Roecklein (301) 443-5970

INSTRUCTION CONCERNING RISK FROM OCCUPATIONAL RADIATION EXPOSURE

A. INTRODUCTION

Section 19.12 of 10 CFR Part 19, "Notices, Instructions and Reports to Workers; Inspections," requires that all individuals working in or frequenting any portion of a restricted area be instructed in the health protection problems associated with exposure to radioactive materials or radiation. This guide describes the instruction that should be provided concerning biological risks to the worker from occupational radiation exposure.

B. DISCUSSION

It is generally accepted by the scientific community that exposure to ionizing radiation may cause biological effects that may be harmful to the exposed organism. These effects are generally classified into two general categories. These categories are Somatic Effects, i.e., effects occurring in the exposed person which, in turn, may be divided into two classes: prompt effects that are observable soon after a large or acute dose (e.g., 25 rems or more in a few hours) and delayed effects such as cancer that may occur years after exposure to radiation; and Genetic Effects,* i.e., abnormalities that may occur in the children of exposed individuals and in subsequent generations. Concerns about these biological effects have resulted in stringent controls on

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*Genetic Effects have not been observed in any of the studies of exposed humans.

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

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FOR COMMENT

doses to individual workers and in efforts to control the collective dose (man-rems) to the worker population.

NRC licensee activities result in a significant fraction of the total occupational radiation exposure in the United States. Regulatory action has recently focused more attention on implementing the philosophy of maintaining occupational radiation exposure at levels that are as low as is reasonably achievable (ALARA). Radiation protection training for all workers who may be exposed to ionizing radiation is an essential component of any program designed to maintain exposure levels ALARA. A clear understanding of what is presently known about the biological risks associated with exposure to radiation will result in more effective radiation protection training and should generate more interest on the part of the worker in minimizing both individual and collective doses. In addition, radiation workers have the right to whatever information on radiation risk is available to enable them to make informed decisions regarding the acceptance of these risks.

At the relatively low levels of occupational radiation exposure in the United States, it is difficult to demonstrate correlations between exposure and effect. There is considerable uncertainty and controversy regarding estimates of radiation risk. In the appendix to this guide, a range of risk estimates is provided (see Table 1). Information on radiation risk has been included from such sources as the 1979 National Academy of Sciences Report of the Committee on the Biological Effects of Ionizing Radiation (BEIR 79),* the International Commission on Radiological Protection (ICRP) Publication 27 entitled "Problems in Developing an Index of Harm," the 1979 report of the science work group of the Interagency Task Force on the Health Effects of Ionizing Radiation, the 1977 report of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR report), and numerous published articles (see the bibliography to the appendix).

C. REGULATORY POSITION

Instruction to workers performed in compliance with §19.12 of 10 CFR Part 19 should be given prior to assignment to work in a restricted area and periodically thereafter. In providing instruction concerning health protection problems associated with exposure to radiation, all workers, including those in supervisory roles, should be given specific instruction on the risk of biological effects resulting from exposure to radiation.

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*The 1979 BEIR report, issued in draft form, is currently being revised. A final version is not yet released but the information from the draft used for this guide is not expected to change significantly.

The instruction should include the information provided in the appendix to this guide and should be presented to all affected workers and supervisors. The information should be discussed during training sessions. Each individual should be given an opportunity to ask questions and should be asked to acknowledge in writing that the instruction has been received.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants regarding the NRC staff's plans for using this regulatory guide.

This proposed guide has been released to encourage public participation in its development. Except in those cases in which a licensee proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the methods to be described in the active guide reflecting public comments will be used in the evaluation of the instructional program for all individuals working in or frequenting any portion of a restricted area and for all supervisory personnel. Implementation by the staff will in no case be earlier than December 1, 1980.

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APPENDIX TO DRAFT REGULATORY GUIDE OH 902-1

INSTRUCTION CONCERNING RISKS FROM OCCUPATIONAL RADIATION EXPOSURE

This instructional material is intended to provide the user with the best available information concerning what is currently known about the health risks from exposure to ionizing radiation. A question and answer format has been used. The questions were developed by the NRC staff in consultation with workers, union representatives, and licensee representatives experienced in radiation protection training. Risk estimates have been compiled from numerous sources generally recognized as reliable. A bibliography is included for the user interested in further study.

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1. What is meant by risk?

Risk can be defined in general as the probability (chance) of injury, illness, or death resulting from some activity. The intent of this document is to estimate and explain the possible risk of injury, illness, or death resulting from occupational radiation exposure.

2. What are the possible health effects of exposure to radiation?

Some of the health effects that exposure to radiation may cause are cancer (including leukemia), birth defects in the children of exposed parents, and cataracts. These effects (with the exception of genetic effects) have been demonstrated in studies of medical radiologists, uranium miners, radium workers, and radiotherapy patients who received excessive doses in the early part of the century. Studies of people exposed to radiation from atomic weapons have also provided data on radiation effects. In addition, radiation effects studies with laboratory animals have provided a large body of data.

The studies mentioned, however, involve levels of radiation exposure that are much higher than those permitted occupationally today. Studies have not shown a clear cause-effect relationship between health effects and current levels of occupational radiation exposure.

3. What is meant by prompt effects, delayed effects, and genetic effects?

Prompt effects are observable shortly after receiving a very large dose in a short period of time. For example, a dose of 450 rems to an average adult will cause vomiting and diarrhea within a few hours; loss of hair, fever, and weight loss within a few weeks; and about a 50 percent chance of death within 1 month without medical treatment. Delayed effects such as cancer and cataracts may occur years after exposure to radiation. Genetic effects occur when there is radiation damage to the genetic material. These effects may show up as birth defects or other conditions in the offspring of the exposed individual and succeeding generations, as demonstrated in animal experiments, although this effect has not been observed in human populations.

4. As nuclear industry workers, which effects should concern us most?

Immediate or prompt effects are very unlikely since large exposures would normally occur only if there were a serious radiation accident. Accident rates in the nuclear industry have been low, and only a few accidents have resulted in overexposures. The probability of serious genetic effects in the children of workers is estimated at about one-third that of other delayed effects. The main concern to industry workers should be the delayed incidence of cancer. The chance of delayed cancer is believed to depend on how much radiation exposure a person gets; therefore, every reasonable effort should be made to keep exposures low.

5. What is the difference between acute and chronic exposure?

Acute radiation exposure, which causes prompt effects and may cause delayed effects, refers to a large dose of radiation received in a short period of time; for example, 450 rems received within a few hours or less. The effects of acute exposures are well known from studies of radiotherapy patients, atomic bomb victims, and accidents that have occurred in nuclear fuel processing. There have been few occupational incidents that have resulted in large acute exposures. Chronic exposure, which may cause delayed effects but not prompt effects, refers to small doses received repeatedly over long time periods, for example, 20-100 mrems (a mrem is one-thousandth of a rem) per week every week for several years. Concern with occupational radiation risk is primarily focused on chronic exposure to low levels of radiation over long time periods.

6. How does radiation cause cancer?

How radiation causes cancer is not well understood. It is impossible to tell whether a given cancer was caused by radiation or by some other of the many apparent causes. However, most diseases are caused by the interaction of several factors. General physical condition, inherited traits, age, sex, and exposure to other cancer-causing agents such as cigarette smoke are a few possible interacting factors. One theory is that radiation activates an existing virus in the body which then attacks normal cells causing them to grow rapidly. Another is that radiation reduces the body's normal resistance to existing viruses which can then multiply and damage cells. Radiation can also damage chromosomes in a cell, and the cell is then directed along abnormal growth patterns. What is known is that, in groups of highly exposed people, a higher than normal incidence of cancer is observed. An increased incidence of cancer has not yet been observed at low radiation levels, although human studies are still incomplete. Higher incidence rates of cancer can be produced in laboratory animals by high levels of radiation.

7. If I receive a radiation dose, does that mean I am certain to get cancer?

Not at all. Everyone gets a radiation dose every day but most people do not get cancer. Even with doses of radiation far above legal limits, most individuals will experience no delayed consequences. There is evidence that the human body will repair some of the damage. The danger from radiation is much like the danger from cigarette smoke. Only a fraction of the people who breathe cigarette smoke get lung cancer, but there is good evidence that smoking increases a person's chances of getting lung cancer. Similarly, there is evidence that large radiation doses increase a person's chances of getting cancer.

Radiation is like most substances that cause cancer in that the effects can be seen clearly only at high doses. Still, it is prudent to assume that smaller doses also have some chance of causing cancer. This is as true for natural cancer-causers such as sunlight and natural radiation as it is for those that are man made such as cigarette smoke, smog, and man-made radiation. As even very small doses may entail some small risk, it follows that no dose should be taken without a reason. Thus, a time-honored principle of radiation protection is to do more than merely meet the allowed regulatory limits; doses should be kept as low as is reasonably achievable (ALARA).

We don't know exactly what the chances are of getting cancer from a radiation dose, but we do have good estimates. The estimates of radiation risks are at least as reliable as estimates for the effects from any other important hazard. Being exposed to typical occupational radiation doses is taking a chance, but that chance is small and reasonably well understood.

It is important to understand the probability factors here. A similar question would be: if you select one card from a full deck, will you get the ace of spades? This question cannot be answered with a simple yes or no. The best answer is that your chances are 1 in 52. However, if 1000 people each select one card from full decks, we can predict that about 20 of them will get an ace of spades. Each person will have 1 chance in 52 of drawing the ace of spades, but there is no way that we can predict which individuals will get the right card. The issue is further complicated by the fact that in 1 drawing by 1000 people, we might get only 15 successes and in another perhaps 25 correct cards in 1000 draws. We can say that if you receive a radiation dose, you will have increased your statistical chances of eventually developing cancer or some other radiation-related injury. The more radiation exposure you get, the more you increase your chances of cancer.

Clearly, there is no simple answer to this question. The best we can do is provide estimates, for large groups, of the increased chances of cancer or other radiation injury resulting from exposure to radiation.

A reasonable comparison involves exposure to the sun's rays. Frequent short exposures provide time for the skin to repair. An acute exposure to the sun can result in painful burning, and excessive exposure has been shown to cause skin cancer. Whether exposure to the sun's rays is short term or spread over time, some of the injury is not repaired and may eventually result in skin cancer.

The effect upon a group of exposed workers may be an increased incidence of cancer over and above the number of cancers that would be expected in that population. Each exposed individual has an increased probability of incurring subsequent cancer. We can say that if 10,000 workers each receive an additional 1 rem in a year, that group is more likely to have a larger incidence of cancer than 10,000 people who do not receive the additional radiation. An estimate of the increased probability of cancer from low radiation doses delivered to large groups is one measure of occupational risk.

8. What are the estimates of the risk of cancer from radiation exposure?

The cancer risk estimates (developed by the organizations identified in Question 9) are presented in Table 1.

TABLE 1

CANCER RISK ESTIMATES FROM EXPOSURE TO LOW-LEVEL RADIATION

Source	Number of Additional Cancers Estimated to Occur in 1 Million People After Exposure of Each to 1 Rem of Radiation
BEIR 1979	268-399
ICRP 1977	300*
UNSCEAR 1977	300*

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*ICRP and UNSCEAR both estimated 100 excess delayed deaths from these 300 radiation-induced cancers. Only about one-third of cancer cases are fatal. Note that the three independent groups are in close agreement on the risk of radiation-induced cancer.

To put these estimates (of Table 1) into perspective, we will use an average of 300 excess cancer cases per million people, each exposed to 1 rem of ionizing radiation. (Most scientists would agree that 300 is a high estimate of risk and may be considered an upper limit.) This means that if in a group of 10,000 workers each receives 1 rem, three would be predicted to develop cancer because of that exposure, although the actual number could be more or less than three (including none).

The American Cancer Society has reported that approximately 25 percent of all adults in the 20-65 year age bracket will develop cancer at some time from all possible causes such as smoking, food, alcohol, drugs, air pollutants, and natural background radiation. Thus in any group of 10,000 workers not exposed to radiation on the job, we can expect about 2,500 to develop cancer. If this entire group of 10,000 workers were to receive an occupational radiation dose of 1 rem each, we could estimate that three additional cases might occur which would give a total of about 2,503. This means that a 1-rem dose to each of 10,000 workers might increase the cancer rate from 25 percent to 25.03 percent, an increase of about 3 hundredths of one percent.

As an individual, if your cumulative occupational radiation dose is 1 rem, your chances of eventually developing cancer during your entire lifetime may have increased from 25 percent to 25.03 percent. If your lifetime occupational dose is 10 rems, we could estimate a 25.3 percent chance of developing cancer.

The normal chance of developing cancer if you receive no occupational radiation dose is about equal to your chance of getting any spade on a single draw from a full deck of playing cards, which is one chance out of four. The additional chance of cancer from an occupational exposure of 1 rem is about equal to your chances of drawing three aces in a row from a deck of cards.

Since cancer resulting from exposure to radiation usually occurs 5 to 25 years after the exposure and since not all cancers are fatal, another useful measure of risk is years of life expectancy lost from a radiation-induced cancer. Several independent studies have indicated that the average loss of life expectancy from exposure to radiation is about 1 day per rem of exposure. In other words, an individual in a population exposed to 1 rem of radiation may on the average lose 1 day of life. The words "on the average" are important, however, because the individual who gets cancer from radiation may lose several years of life expectancy while his more fortunate coworkers suffer no loss. The International Commission on Radiological Protection (ICRP) estimated that the average number of years of life lost from a fatal industrial accident is 30 while the average number of years of life lost from a fatal radiation-induced cancer is 10.

It is important to realize that these risk numbers are only estimates. Many difficulties are involved in designing research studies that can accurately measure the small increases in cancer incidence due to low exposures to radiation as compared to the normal incidence of cancer. There is still uncertainty and a great deal of controversy with regard to estimates of radiation risk. The numbers used here result from studies involving high doses and high dose rates, and they may not apply to doses at the lower occupational levels of exposure. At low dose levels, it is possible that the risk could be zero. The NRC and other agencies both in the United States and abroad are continuing extensive long-range research programs on radiation risk.

The National Academy of Sciences Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR) and others feel that these risk estimates are higher than would actually occur and represent an upper limit on the risk. However, they are considered by the NRC staff to be the best available estimates that the worker can use to make an informed decision concerning acceptance of the risks associated with exposure to radiation. Although the estimated increased risks of cancer are relatively low, there is a chance that they are not zero. A worker who decides to accept this small increased risk should make every effort to keep exposure to radiation as low as is reasonably achievable to avoid unnecessary risk.

9. What groups of expert scientists have studied the risk from exposure to radiation?

Since 1956, the National Academy of Sciences established two advisory committees to consider radiation risks. The first of these was the Advisory Committee on the Biological Effects of Atomic Radiations (BEAR) and more recently it was renamed the Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR). These committees have periodically reviewed the extensive research being done on the health effects of ionizing radiation and have published estimates of the risk of cancer from exposure to radiation (1972 and 1979* BEIR reports). The International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurement (NCRP) are two groups of renowned scientists who have studied radiation effects and published risk estimates (ICRP Publication 26, 1977). In addition, the United Nations established an independent study group that published an extensive report in 1977, including estimates of cancer risk from ionizing radiation (UNSCEAR 1977).

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* The draft publication of the 1979 BEIR report is currently under revision. However, the risk estimates are not expected to change significantly.

10. Can a worker become sterile or impotent from occupational radiation exposure?

Observation of radiation therapy patients who receive localized exposures, usually spread over a few weeks, has shown that a dose of 500-800 rems to the gonads can produce permanent sterility in males or females (an acute whole-body dose of this magnitude would probably result in death within 30 days). An acute dose of 20 rems to the testes can result in a measurable but temporary reduction in sperm count. Such high exposures on the job could result only from serious and unlikely radiation accidents. The whole-body dose required to make someone impotent is also greater than the lethal dose. Thus, exposure to permitted occupational levels of radiation has no observed effect on fertility and should have no physical effect on the ability to function sexually.

11. How can we compare radiation risk to other kinds of health risks?

Perhaps the most useful unit for comparison among health risks is the average number of days of life expectancy lost per unit of exposure to each particular health risk. Estimates are calculated by looking at a large number of individuals, recording the age when death occurs from apparent causes, and estimating the number of days of life lost as a result of these early deaths. The total number of days of life lost is then averaged over the total group observed.

Several studies have compared the projected loss of life expectancy resulting from exposure to radiation with other health risks. Some representative numbers are presented in Table 2.

TABLE 2

ESTIMATED LOSS OF LIFE EXPECTANCY FROM HEALTH RISKS

Health Risk	Estimates of Days of Life Expectancy Lost, Average
Smoking 20 cigarettes/day	2370 (6.5 years)
Overweight (by 20%)	985 (2.7 years)
All accidents combined	435 (1.2 years)
Auto accidents	200
Alcohol consumption (U.S. average)	130
Home accidents	95
Drowning	41
Safest jobs (such as teaching)	30
Natural background radiation, calculated	8
Medical X-rays (U.S. average), calculated	6
All catastrophes (earthquake, etc.)	3.5
1 rem occupational radiation dose, calculated (industry average is 0.34 rem/yr)	1
1 rem/yr for 30 years, calculated	30
5 rems/yr for 30 years, calculated	150

These estimates indicate that the health risks from occupational radiation exposure are not greater than the risks associated with many other events or activities we encounter in normal day-to-day activities.

A second useful comparison is to look at estimates of the average number of days of life expectancy lost from exposure to radiation and from common industrial accidents at radiation-related facilities and to compare this number with days lost from other occupational accidents. Table 3 shows average days of life expectancy lost as a result of fatal work-related accidents. Note that the data for occupations other than radiation related do not include death risks from other possibly related hazards such as exposure to toxic chemicals, dusts, or unusual temperatures. Note also that occupational exposure at the 5 rems per year limit for 50 years, though highly unlikely, may result in a risk comparable to mining and heavy construction, using high-risk estimates.

TABLE 3

ESTIMATED LOSS OF LIFE EXPECTANCY FROM INDUSTRIAL ACCIDENTS*

Industry Type	Estimates of Days of Life Expectancy Lost, Average
All industry	74
Trade	30
Manufacturing	43
Service	47
Government	55
Transportation and utilities	164
Agriculture	277
Construction	302
Mining and quarrying	328
Radiation accidents, death from exposure	<1
Radiation dose of 0.5 rem/yr, 50 years, calculated	25
Radiation dose of 5 rems/yr, 50 years	250
Industrial accidents at nuclear facilities (nonradiation)	58

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* Adapted from Cohen and Lee, A Catalogue of Risk and Health Implications of Nuclear Power Production, World Health Organization.

Industrial accident rates in the nuclear industry and related occupational areas have been relatively low during the entire history of the industry (see Table 4). This is due perhaps to the early and continuing emphasis on tight safety controls. The relative safety of various occupational areas can be seen by comparing the probability of accidental death per 10,000 workers over a 40-year working lifetime. These figures do not include death from possible causes such as exposure to toxic chemicals or radiation.

TABLE 4

PROBABILITY OF ACCIDENTAL DEATH BY TYPE OF OCCUPATION*

Occupation	Number of Accidental Deaths for 10,000 Workers for 40 Years
Mining	252
Construction	228
Agriculture	216
Transportation and public utilities	116
All industries	56
Government	44
Nuclear industry (1975 data)	40
Manufacturing	36
Services	28
Wholesale and trade	24

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*Adapted from Accident Facts, National Safety Council, 1979, and Operational Accidents and Radiation Exposure Experience, WASH-1192, Atomic Energy Commission, 1975.

12. What are the NRC radiation dose limits?

Federal regulations currently limit occupational radiation dose to 1-1/4 rems in any calendar quarter or specified 3-month period. However, when there is documented evidence that a worker's previous occupational dose is low enough, a licensee may permit a dose of up to 3 rems per quarter or 12 rems per year. The accumulated dose may not exceed 5(N - 18) rems where N is the individual's age in years, i.e., the lifetime occupational dose may not exceed an average of 5 rems for each year above the age of 18.

13. What is meant by ALARA?

In addition to providing an upper limit on an individual's permissible radiation exposure, the NRC also requires that its licensees maintain exposures as far below the limit as is reasonably achievable (ALARA). This means that every activity at a nuclear facility involving exposure to radiation should be planned so as to minimize unnecessary exposure to individual workers and also to the worker population. A job that involves exposure to radiation should be done only when it is clear that the benefit justifies the risks assumed. All design, construction, and operating procedures should be reviewed with the objective of reducing unnecessary exposures.

14. Has the ALARA concept been applied if, instead of reaching dose limits during the first week of a quarter, the worker's dose is spread out over the whole quarter?

No. At low doses the health effects do not seem to be affected by dose rate. The risk of cancer from low doses is considered to be proportional to the amount of exposure, not the rate at which it is received. Spreading the dose out over time or over larger numbers of people does not reduce the overall risk. The ALARA concept has been followed only when the collective dose is reduced by reducing the time of exposure or decreasing radiation levels in the working environment.

15. What is meant by collective dose and why should it be maintained ALARA?

Nuclear industry activities expose an increasing number of people to occupational radiation in addition to the radiation doses they receive from natural background radiation and medical radiation exposures. The collective occupational dose (man-rems) is the sum of all occupational radiation exposure received by all the workers in an entire worker population. For example, if 100 workers each receive 2 rems, the individual dose is 2 rems and the collective dose is 200 man-rems. The total additional risk of cancer and genetic effects in an exposed population is assumed to depend on the collective dose.

It should be noted that, from the viewpoint of risk to a total population, it is the collective dose that must be controlled. For a given collective dose, the number of health effects is believed to be the same even if a larger number of people share the dose. Therefore, spreading the dose out may reduce the individual risk, but not that of the population.

Efforts should be made to maintain the collective dose ALARA so as not to unnecessarily increase the overall population incidence of cancer and genetic effects.

16. Is the use of extra workers a good way to reduce risks?

There is a "yes" answer to this question and a "no" answer. For a given job involving exposure to radiation, the more people who share the work, the lower the average dose to an individual. The lower the dose, the lower the risk. So, for you as an individual, the answer is "yes."

But how about the risk to the entire group of workers? The risk of cancer depends on the total amount of radiation energy absorbed by human tissue, not on the number of people to whom this tissue belongs. Therefore, if 30 workers are used to do a job instead of 10, and if both groups get the same collective dose (man-rems), the total cancer risk is the same, and nothing was gained for the group by using 30 workers. From this viewpoint the answer is "no." The risk was not reduced but simply spread around among a larger number of individuals.

Unfortunately, spreading the risk around often results in a larger collective dose for the job. Workers are exposed as they approach a job, while they are getting oriented to do the job, and as they withdraw from the job. The dose received during these actions is called nonproductive. If several crew changes are required, the nonproductive dose can become very large. Thus it can be seen that the use of extra workers may actually increase the total occupational dose and the resulting risks.

The use of extra workers to comply with NRC dose limits is not the way to reduce the risk of radiation-induced cancer for the worker population. At best, the total risk remains the same, and it may even be increased. The only way to reduce the risk is to reduce the collective dose; that can be done only by reducing the radiation levels, the working times, or both.

17. Why doesn't the NRC impose collective dose limits?

Compliance with individual dose limits can be achieved simply by using extra workers. However, compliance with a collective dose limit (such as 100 man-rems per year for a licensee) would require reduction of radiation levels, working times, or both. But there are many problems associated with setting appropriate collective dose limits.

For example, we might consider applying a single collective dose limit to all licensees. The selection of such a collective dose limit would be almost impossible because of the large variations in collective doses among licensees. A power reactor could reasonably be expected to have an average annual collective dose of several hundred man-rems. However, a small radiography licensee could very well have a collective dose of only a few man-rems in a year.

Even choosing a collective dose limit for a group of similar licensees would be almost as difficult. Radiography licensees as a group had an average collective dose in 1977 of 9 man-rems. However, the smallest collective dose for a radiography licensee was less than 1 man-rem, and the largest was 401 man-rems.

Setting a reasonable collective dose limit for each individual licensee would also be very difficult. It would require a record of all past collective doses on which to base such limits. Setting an annual collective dose limit would then amount to an attempt to predict a reasonable collective dose for each future year. In order to do this, it would be necessary to be able to predict changes in each licensed activity that would increase or decrease the collective dose. In addition, annual collective doses vary significantly from year to year according to the kind and amount of maintenance required, which cannot generally be predicted in advance. Following all such changes and revising limits up and down would be very difficult if not impossible. However, these efforts would be necessary if a collective dose limit were to be reasonable and help minimize doses and risks.

18. How are radiation dose limits established?

The NRC establishes occupational radiation dose limits based on guidance to Federal agencies from the Environmental Protection Agency (EPA) and on NCRP and ICRP recommendations. Scientific reviews of research data on biological effects such as the BEIR report are also considered.

19. What are the typical radiation doses received by workers?

The NRC requires that certain categories of licensees report data on annual worker doses and doses for all workers who terminate employment with licensees.

Data were received on the occupational doses in 1977 of approximately 100,000 workers in power reactors, industrial radiography, fuel processing and fabrication facilities, and manufacturing and distribution facilities. Of this total group, 85 percent received an annual dose of less than 1 rem according to these reports; 95 percent received less than 2 rems; fewer than 1 percent exceeded 5 rems in any 1 year. The average annual dose of these workers who were monitored and had measurable exposures is about 0.65 rem. A study completed by the EPA, using 1975 exposure data for 1,260,000 workers, indicated that the average annual dose for all workers who received a measurable dose was 0.34 rem.

20. What happens if a worker exceeds the quarterly exposure limit?

Radiation protection limits, such as 3 rems in 3 months, are not absolute limits below which it is safe and above which there is danger. Exceeding a limit does not imply that you have suffered an injury. A good comparison is with the highway speed limit which is selected to limit accident risk and still allow you to get somewhere. If you drive at 75 mph, you increase your risk of an auto accident to levels that are not considered acceptable by the people who set speed limits, even though you may not actually have an accident. If a worker's radiation dose repeatedly exceeds 3 rems in a quarter, the risk of health effects could eventually increase to a level that is not considered acceptable to the NRC. Exceeding an NRC protection limit does not necessarily mean that any adverse health effects are going to occur. It does mean that a licensee's safety program has failed in some respect and that the NRC and the licensee should investigate to make sure the problems are corrected.

If an overexposure occurs, the regulations prohibit any additional occupational exposure to that individual during the calendar quarter. The licensee is required to file a report to the NRC and may possibly be subject to a fine, just as you are subject to a traffic fine for exceeding the speed limit. In both cases, the fines and, in some serious or repetitive cases, suspension of license are intended to encourage efforts to operate within the limits. The safest limits would be 0 mph and 0 rem per quarter. But then we wouldn't get anywhere.

21. Why do some facilities establish administrative limits that are below the NRC limits?

There are two reasons. First, paragraph 20.1(c) of the NRC regulations states that licensees should keep exposures to radiation ALARA. By requiring specific approval for worker doses in excess of set levels, more careful risk-benefit analysis can be made as each additional increment of dose is approved for a worker. Secondly, a facility administrative limit that is set lower than the quarterly NRC limit provides a safety margin designed to help the licensee avoid overexposures.

22. Several scientists have recently suggested that NRC limits are too high and should be lowered. What are the arguments for lowering the limits?

In general, those critical of present dose limits say that the individual risk is higher than estimated by the BEIR Committee and the ICRP. A few studies have indicated that a given dose of radiation may be more likely to cause biological effects than previously thought. The controversy is focused on studies involving groups of exposed individuals. Opinions differ on the validity of the research methods used and the methods of statistical analysis. The chief problem is that, with small groups, the incidence of effects such as leukemia is small. It cannot be shown without question that these effects were more frequent in the exposed study group than in the unexposed group used for comparison or that any observed effects were caused by the exposure to radiation.

The current BEIR committee concluded that claims of higher risk had "no substance," and nearly one-half of the committee members were convinced that the BEIR risk estimates were actually too high. The NRC staff is committed to a continuing review of research on radiation risk and is funding a study to design new research on human effects from exposure to radiation.

23. What are the arguments against lowering the NRC dose limits?

The estimated health risks associated with current average occupational radiation doses (e.g., 0.5 rem/yr for 50 years) are comparable to or less than risk levels in other occupational areas considered to be among the safest.

Exposure to 5 rems/yr for 50 years, which virtually never occurs, would increase the estimated risk to levels comparable to risks in mining and heavy construction. If the dose limits were lowered significantly, the number of people required to complete many jobs would increase. The collective dose would then increase since more individuals would be receiving nonproductive exposure while entering and leaving the work area and preparing for the job. The total number of health effects might go up as the collective dose increased.

The regulatory standards for dose limits are based on the recommendations of the Federal Radiation Council, the NCRP, and the ICRP. At the time these standards were developed, about 1960, it was considered unlikely that exposure of these levels during a working lifetime would result in clinical evidence of injury or disease different from that occurring in the unexposed population. The scientific data base for the standards consisted primarily of human experience (X-ray exposures to medical practitioners and patients, ingestion of radium by watch dial painters, early effects observed in Japanese atomic bomb survivors, radon exposures of uranium miners, occupational radiation accidents) involving very large doses delivered at very high dose rates. The data base also included the results of a large number of animal experiments involving high doses and dose rates. The animal experiments were particularly useful in the evaluation of genetic effects. The observed effects were related to low-level radiation through a linear, nonthreshold extrapolation procedure. Based on this approach, the regulations in 10 CFR Part 20, "Standards for Protection Against Radiation," also state that licensees should maintain all radiation exposures, and releases of radioactive materials in effluents, as low as is reasonably achievable.

Reducing the dose limits, for example, by a factor of 10 (that is, from 5.0 rems/yr to 0.5 rem/yr) has been analyzed by the NRC staff. An estimated 2.6 million man-rems could be saved from 1980 through the year 2000 by nuclear power plant licensees if compliance with the new limit was achieved by lowering the radiation levels, working times, or both, rather than by using extra workers. It is estimated that something like $23 billion would be spent toward this purpose. Spending $23 billion to save 2.6 million man-rems would amount to spending $30 to $90 million to prevent each potential radiation-induced cancer death. Society may consider this cost unacceptably high for individual protection.

24. Are there any areas of concern about radiation risks that might result in lowering the NRC dose limits?

Three areas of concern to the NRC staff are specifically identified below:

a. An independent study has indicated that a given dose of neutron radiation is more likely to cause biological effects than previously thought. Although the scientific community has not yet agreed with the results of this study, workers should be advised of the possibility of higher risk when entering areas where exposure to neutrons will occur.

b. It has been known for some time that rapidly growing living tissue is more sensitive to injury from radiation than tissue in which the cells are not reproducing rapidly. Thus the unborn embryo or fetus is more sensitive to radiation injury than an adult. The NCRP recommended in Report No. 39 that special precautions be taken when an occupationally exposed woman could be pregnant in order to protect the embryo or fetus. In 1975, the NRC issued Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure," in which it is recommended that licensees instruct all workers concerning this special risk. The guide recommends that all ers be advised that the NCRP recommended the maximum permissible dose to the embryo or fetus from occupational exposure of the mother should not exceed 0.5 rem for the full 9-month pregnancy period. In addition, the guide suggests options available to the female employee who chooses not to expose her unborn child to this additional risk.

c. Also of special interest is the indication that female workers are subject to more risk than male workers. In terms of all types of cancer except leukemia, the 1979 BEIR analysis indicates that female workers have a risk of developing radiation-induced cancer that is approximately one and one-half times that for males. Incidence of radiation-induced leukemia is about the same for both sexes. Female workers should consider carefully this difference in the risks of radiation-induced cancer in deciding whether or not to seek work involving exposure to radiation.

25. How much radiation does the average person who does not work in the nuclear industry receive?

We are all exposed from the moment of conception to ionizing radiation from several sources. Our environment, and even the human body, contains naturally occurring radioactive materials that contribute some of the background radiation we receive. Cosmic radiation originating in space and in the sun contributes additional exposure. The use of X-rays and radioisotopes in medicine and dentistry adds considerably to our population exposure.

Table 5 shows estimated average individual exposure in millirems from natural background and other sources.

TABLE 5

U.S. GENERAL POPULATION EXPOSURE ESTIMATES (1978)*

Source	Average Individual Dose (mrem/yr)
Natural background	100
Release of radioactive material by mining, milling, etc.	5
Medical	90
Nuclear weapons development (primarily fallout)	5-8
Nuclear energy	0.28
Consumer products	0.03
Total	~ 200 mrem/yr

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*Adapted from a report by the Interagency Task Force on the Health Effects of Ionizing Radiation published by the Department of Health, Education, and Welfare.

Thus, the average individual in the general population receives about 0.2 rem of radiation exposure each year from sources that are a part of our natural and man-made environment. By the age of 20 years, an individual has accumulated about 4 rems. The most likely target for reduction of population exposure is medical uses.

26. Why aren't medical exposures considered as part of a workers allowed dose?

Equal doses of medical and occupational radiation have equal risks. Medical exposure to radiation should be justified for reasons quite different, however, from those applicable to occupational exposure. A physician prescribing an X-ray should be convinced that the benefit of the resulting medical information justifies the risk associated with the radiation. Each worker must decide on the acceptance of occupational radiation risk just as each worker must decide on the acceptability of any other occupational hazard.

For another point of view, consider a worker who receives a series of X-rays or a radiopharmaceutical in connection with an injury or illness resulting in a dose of 2 rems. This dose and implied risk should be justified on medical grounds. If the worker had also received 2 rems of dose on the job, the combined dose of 4 rems would not incapacitate the worker. Restricting the worker from additional job exposure during the quarter would have no effect one way or the other on the risk from the 2 rems already received from medical exposure. If the individual worker accepts the risks associated with the X-rays on the basis of the medical benefits and the risks associated with job-related exposure on the basis of employment benefits, it would be inequitable to restrict the individual from employment in restricted areas for the remainder of the quarter.

27. What is meant by internal exposure?

Internal exposure to radiation results when radioactive materials are taken into the body by breathing, ingestion, or absorption through the skin. Different types of material locate for a period of time in different parts of the body or pass through the body, resulting in some dose to the exposed tissues.

Internal exposure can be estimated by measuring the radiation emitted from the body or by measuring the radioactive materials contained in biological samples such as urine or feces. Dose estimates can also be made if one knows how much radioactive material is in the air and the length of time during which the air was breathed.

28. How are the limits for internal exposure set?

Calculations are made to determine the quantity of radioactive material that has been taken into the body and the total organ dose that would result. Then, based on limits established for particular body organs similar to 1-1/4 rems in a calendar quarter for whole-body exposure, the regulations specify maximum permissible concentrations of radioactive material in the air to which a worker can be exposed for 40 hours per week. The regulations also require that efforts be made to keep internal exposure ALARA.

Internal exposure is controlled by limiting the release of radioactive material into the air and by carefully monitoring the work area for airborne radioactivity and surface contamination. Protective clothing and respiratory (breathing) protection may be used whenever the possibility of contact with loose radioactive material cannot be prevented.

29. Is the dose an individual received from internal exposure added to that received from external exposure?

Exposure to radiation that results from radioactive materials taken into the body is measured, recorded, and reported to the worker separately from external dose. The internal dose to the whole body or to specific organs does not at this time count against the 3 rems per calendar quarter limit. ICRP recommendations are that the internal and external doses should be summed. This recommendation is under study by the staffs of the NRC and the EPA.

30. How is a worker's radiation dose determined?

A worker may wear two types of radiation-measuring devices. A self-reading pocket dosimeter records the exposure to incident radiation and can be read out immediately upon finishing a job involving external exposure to radiation. A film badge or TLD badge records radiation dose, either by the amount of darkening of the film or by storing energy in the TLD crystal. Both these devices require processing to determine the dose and are considered more reliable than the pocket dosimeter. A worker's official report of dose received is normally based on film or TLD badge readings.

31. What are my options if I decide not to accept the risks associated with occupational radiation exposure?

If the risks from exposure to radiation that may be expected to occur during your work are unacceptable to you, you could request a transfer to a job that does not involve exposure to radiation. However, the risks associated with exposure to radiation that workers, on the average, actually receive are considered acceptable, compared to other occupational risks, by virtually all the scientific groups that have studied them. Thus, your employer is not obligated to guarantee you a transfer if you decide not to accept an assignment requiring exposure to radiation.

You also have the option of seeking other employment in a nonradiation occupation. However, the studies that have compared occupational risks in the nuclear industry to those in other job areas indicate that nuclear work is relatively safe. Thus, you will not necessarily find significantly lower risks in another job.

A third option would be to practice the most effective work procedures so as to keep your exposure ALARA. Be aware that reducing time of exposure, maintaining distance from radiation sources, and using shielding can all lower your exposure. Plan radiation jobs carefully to increase efficiency while in the radiation area. Learn the most effective methods of using protective clothing to avoid contamination. Discuss your job with the radiation protection personnel who can suggest additional ways to reduce your exposure.

32. Where can I get additional information on radiation risk?

The following list suggests sources of useful information on radiation risk:

Your Employer

The radiation protection or health physics office in the facility where you are employed.

Nuclear Regulatory Commission

Address: Occupational Health Standards Branch
Office of Standards Development
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Phone: 301-443-5970

NRC Regional Offices

King of Prussia, PA 19406	215-337-5000
Atlanta, GA 30303	404-221-4503
Glen Ellyn, IL 60137	312-932-2500
Arlington, TX 76012	817-334-2841
Walnut Creek, CA 94596	415-943-3700

Department of Health, Education, and Welfare

Address: Office of Public Affairs
Bureau of Radiological Health
Department of Health, Education, and Welfare
Room 15-B-42, HF1-40
5600 Fishers Lane
Rockville, MD 20857

Phone: 301-443-3285

Environmental Protection Agency

Address: Office of Radiation Programs
U.S. Environmental Protection Agency
401 M Street, SW
Washington, D.C. 20460

Phone: 703-557-9710

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